UNITED STATES
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MEMORANDUM TO: Jeffrey A. Clark, Deputy Director
Division of Reactor Safety
Region IV

FROM: Kathryn M. Brock, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: RESPONSE TO TASK INTERFACE AGREEMENT 2014-10 RELATED TO THE REGULATORY POSITION ON EMERGENCY DIESEL GENERATOR MISSION TIME FOR OPERABILITY EVALUATIONS AT CALLAWAY PLANT, UNIT NO. 1 (CAC NO. MF5099, EPID L-2015-LRA-0001)

By memorandum dated March 30, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15089A455), the U.S. Nuclear Regulatory Commission (NRC) Region IV (RIV), Division of Reactor Safety, requested assistance from the NRC Office of Nuclear Reactor Regulation (NRR) to clarify the NRC staff's position regarding Union Electric Company's (Ameren Missouri, the licensee) assumption of a 7-day mission time for the emergency diesel generator (EDG) in operability determinations at Callaway Plant, Unit No. 1 (Callaway).

The NRC staff has reviewed the matter and has determined that, although "EDG mission time" is not a term used in Callaway's licensing basis documents, the Callaway Final Safety Analysis Report does assume in designing for and analyzing for a design basis accident that all offsite power is restored within 7 days. The basis for this determination and the remaining answers to the Task Interface Agreement (TIA) questions are provided in the enclosed TIA response. The conclusions in this TIA response represent the NRC staff's position based on the licensing basis information specific to Callaway. The conclusions presented in this TIA response are not generic and do not directly apply to other licensees or sites.

Enclosure:
TIA Response

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RESPONSE TO TASK INTERFACE AGREEMENT 2014-10

REGULATORY POSITION ON EMERGENCY DIESEL GENERATOR

MISSION TIME FOR OPERABILITY EVALUATIONS

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By memorandum dated March 30, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15089A455), the U.S. Nuclear Regulatory Commission (NRC) Region IV (RIV), Division of Reactor Safety, requested technical assistance from the NRC Office of Nuclear Reactor Regulation (NRR) to clarify the NRC staff position regarding Union Electric Company's (Ameren Missouri, the licensee) assumption of a 7-day mission time for the emergency diesel generator (EDG) in operability determinations at the Callaway Plant, Unit No. 1 (Callaway).

Specifically, Task Interface Agreement (TIA) 2014-10 requested answers to the following questions:

1. What is the mission time and basis for the EDGs at the Callaway Plant?

2. How would the mission time differ absent the [Final Safety Analysis Report (FSAR)] Section 3.1.2 assumption?

3. To what extent should the licensee have pre-established agreements, contracts, and procedures, to ensure that offsite power is restored in 7 days?

4. To what extent should non-safety, non-seismic Category 1 equipment (offsite power distribution and switchyard components) assumed for accident mitigation be included in the scope of equipment included in monitoring the effectiveness of maintenance as required by [Title 10 of the *Code of Federal Regulations* (10 CFR)] 50.65?

5. What types of operator actions are allowed and are consistent with safety-related equipment mission time?

2.0 BACKGROUND

EDGs supply a highly reliable self-contained onsite standby source of alternating current (AC) power in the event of a complete loss-of-offsite power (LOOP). Each diesel generator is rated at 6,201 kiloWatt (kW) for continuous operation. Additional ratings are 6,635 kW for 2,000 hours, 6,821 kW for 7 days, and 7,441 kW for 30 minutes. The generator 2-hour rating is equal to the 7-day rating. The EDGs are designed to provide sufficient power for the electrical

loads required for a safe shutdown of the plant. This includes the loads required to mitigate the effects of a design-basis accident (DBA) with a complete LOOP plus a single failure in the onsite power system. The technical specifications (TS) required testing is performed in accordance with NRC Regulatory Guide (RG) 1.9, Revision 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," July 1993 (ADAMS Accession No. ML003739929). The testing demonstrates continuing capability of the EDGs to mitigate the consequences of an accident. During a LOOP, some of the loads typically supplied by the EDGs are pumps that circulate the necessary water to indirectly cool the water that cools the core and prevents overheating of the fuel cladding fission product barrier.

Callaway has two redundant and independent 4.16-kV Class 1E buses (divisions), each capable of being powered from one of the two offsite power circuits or an onsite EDG. Redundant safety related loads can be automatically connected to the two electrical divisions such that in the event of a single failure, the operation of one of the two divisions is available to meet minimum safety requirements.

During Callaway refueling outage 19, the licensee assessed and documented the operability of EDG A jacket water pump and EDG B intercooler pump mechanical seal leakage in Callaway Action Requests (CARs) 2013-03303 and 2013-03613, respectively. The licensee concluded that EDG A and EDG B were operable because the leakage rates were less than that which would have caused their expansion tanks to fall below a minimum level based upon a 7-day mission time. Subsequently, the licensee transitioned from Mode 5 to Mode 4 on May 22, 2013, and continued to Mode 1 on May 27, 2013.

The RIV staff reviewed the licensee's evaluations and believed that the conclusion that the EDGs were operable was incorrect because the jacket water expansion tank level would have fallen below minimum over a 30-day mission time. However, the RIV staff concluded that since the licensee had procedural guidance for refilling the EDG jacket water expansion tank from a safety-related source, the licensee could have maintained continuous operation for 30 days or more despite the leakage and, therefore, these evaluations did not represent immediate safety concerns.

Licensee Position

The licensee's operability determinations, CARs 2013-03303 and 2013-03613, were based upon on a 7-day mission time. The licensee contends (ADAMS Accession No. ML15089A455) that the Callaway FSAR Section 3.1.2 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, "Electric power systems," provide the basis for the restoration of all offsite power in 7 days following a LOOP/loss-of-coolant accident (LOCA) DBA and that, therefore, 7 days was appropriate for use as a mission time.

The Callaway FSAR Section 3.1.2 states, in part, that in designing for and analyzing for a DBA, the assumption was made that all offsite power is simultaneously lost and is restored within 7 days (ADAMS Accession No. ML17067A360). General Design Criterion 17 states, in part, that electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits, each of which shall be designed to be available in sufficient time following a loss of all onsite AC power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The licensee stated that the Callaway licensing basis was based on RG 1.9, Revision 3, which does not include the

example included in RG 1.9, Revision 4, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," March 2007 (ADAMS Accession No. ML070380553), that for an extended LOOP, a 30-day period should be considered with refueling every 7 days.

Therefore, the licensee believes that the mission time for the EDGs at Callaway is 7 days and that for the remaining 23 days analyzed in the FSAR, offsite power is assumed to be available to the station. The licensee stated that changing the mission time of the EDGs from 7 days to 30 days would be a change to the Callaway licensing basis. The licensee also stated that mission time refers to the amount of time equipment must be able to run in a "hands off" condition (i.e., no operator actions allowed except for those specifically authorized by the FSAR).

Region IV Staff Position

The RIV staff based the determination of a 30-day mission time on the Callaway FSAR Chapter 15 accident analysis, which credits only seismic Category I, Class 1E, and Institute of Electrical and Electronics Engineers (IEEE) qualified equipment, instrumentation, and components in the mitigation of the consequences of Conditions II, III, and IV events. Since offsite power does not meet these requirements, the RIV staff position is that the onsite power system should be credited for mitigating the consequences of a postulated DBA and anticipated operational occurrences. Therefore, because the Chapter 15 accident analyses continued to a 30-day evaluation period for the worst case, the RIV staff inferred that the EDGs must be capable of providing emergency power for the duration of the assumed accident. This resulted in the RIV staff determination of a 30-day mission time for the EDGs.

The RIV staff questioned the additional single failure assumption listed in Callaway FSAR Section 3.1.2e for the restoration of all offsite power within 7 days as adequate justification supporting the licensee's position of a 7-day EDG mission time.

3.0 REGULATORY REQUIREMENTS

The regulation at 10 CFR 50.36 requires, in part, that each applicant for a license authorizing operation of a utilization facility include in the application proposed TS in accordance with the requirements of this section. The TS, when approved, become part of the license for the facility.

The regulation at 10 CFR 50.36(c) further requires, in part, that the TS include limiting conditions for operation. Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TS until the condition can be met. The TS require that specific structures, systems, and components (SSCs) be operable given the plant condition (operational mode).

The regulation at 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS).
Section 50.46(b)(5), "Long-term cooling," states "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The regulation at 10 CFR 50.63 requires, in part, that each nuclear power plant licensed to operate be able to withstand for a specified duration and recover from a station blackout. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency AC power sources; (ii) The reliability of the onsite emergency AC power sources; (iii) The expected frequency of loss of offsite power; and (iv) The probable time needed to restore offsite power.

The regulation at 10 CFR 50.65 requires the monitoring of the effectiveness of maintenance at nuclear power plants during all conditions of plant operation.

The regulation at 10 CFR 50.65(a)(1) further requires that each holder of an operating license monitor the performance or condition of SSCs against licensee-established goals in a manner sufficient to provide reasonable assurance that the SSCs are capable of performing their intended functions. Per 10 CFR 50.65(a)(2), monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function.

The regulation at 10 CFR 50.65(b)(2) states that the scope of the monitoring program specified in 10 CFR 50.65(a)(1) shall include non-safety related SSCs meeting criteria as listed in 10 CFR 50.65(b)(2)(i)-(iii). With regard to offsite power distribution and switchyard components, which are non-safety related SSCs, the applicable criterion at Callaway is 10 CFR 50.65(b)(2)(iii), non-safety related SSCs whose failure could cause a reactor scram or actuation of a safety-related system.

The regulation at 10 CFR Part 50, Appendix A, GDC 17, "Electric power systems," requires, in part, an onsite electric power system and an offsite electric power system to permit functioning of SSCs important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. As such, either one of the power sources (assuming the other is not available) is required to operate for the duration of anticipated operational occurrences and postulated accidents to support operation of safety-related equipment.

The regulation at 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," states that:

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Applicable Regulatory Guidance

The Callaway FSAR, Appendix 3A, "Conformance to NRC Regulatory Guides" states, in part:

> With respect to the original selection, design and qualification of emergency diesel generators (per Revision 1 of [RG 1.9]), the recommendations of this regulatory guide were met. With regard to periodic, in-service testing of the diesel generators per Revision 3 of this regulatory guide, testing is performed in accordance with the plant Technical Specifications. The testing requirements in the Technical Specifications are based on Regulatory Guide 1.9, Revision 3.

Revision 1 of RG 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," November 1978 (ADAMS Accession No. ML13226A211), applies to Callaway for the original selection, design, and qualification of EDGs.

Per Appendix 3A of the Callaway FSAR, Callaway follows RG 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 1997 (ADAMS Accession No. ML003761662), as a method for complying with the provisions of 10 CFR 50.65. Revision 2 of RG 1.160 endorses Nuclear Utility Management and Resources Council (NUMARC) 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 1996 (ADAMS Accession No. ML101020415). Section 8.2.1.5 of NUMARC 93-01, Revision 2, provides guidance on scoping equipment into the maintenance rule under 10 CFR 50.65(b)(2)(iii).

4.0 EVALUATION

The following evaluation provides the basis and related information developed for responding to the specific questions posed by the RIV staff.

1. What is the mission time and basis for the EDGs at the Callaway Plant?

The term "mission time" is not used in the Callaway licensing basis. In the NRC licensing documents and guidance, the term "mission time" is used in many different contexts, and according to its context can assume different values. In the context of operability and a deterministic evaluation of overall plant safety, Inspection Manual Chapter (IMC) 0326, "Operability Determinations & Functionality Assessments for Conditions Adverse to Quality or Safety," dated January 31, 2014 (ADAMS Accession No. ML13274A578), states that "the mission time is the duration of SSC operation that is credited in the design basis for the SSC to perform its specified safety function."

Chapter 3 of the Callaway FSAR describes the design of structures, components, equipment, and systems which are necessary to assure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safe shutdown condition.

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline values of 10 CFR Part 100, "Reactor Site Criteria."

Callaway FSAR Section 3.1 discusses the extent to which the design criteria for Standardized Nuclear Unit Power Plant System SSCs important to safety comply with the GDC.

In general, the failure of one onsite safety related power source or one protection train is considered the limiting single failure. Callaway FSAR Table 15.0-7, "Single Failures Assumed in Accident Analyses" (ADAMS Accession No. ML17065A090), further states that the licensee has assumed the loss of one protection train as the limiting case for postulated events and accident conditions.

Therefore, in the context of the Callaway licensing basis, EDG mission time refers to the amount of time the EDG is required to operate to supply power to safety systems that mitigate the effects of accidents and events delineated in the safety analysis and to power the equipment necessary for long term core cooling.

Callaway FSAR Section 3.1.2, "Additional Single Failure Assumptions," state, in part, the following:

> In designing for and analyzing for DBAs (i.e., large break loss-of-coolant accident (LBLOCA), main steam line break, main feedwater line break, rod injection, locked RCP rotor or RCP shaft break, fuel handling accident, or steam generator tube rupture), the following assumptions (a-f) are made in addition to postulating the initiating event. In designing for and analyzing for an ANS Condition III small break loss-of-coolant accident (SBLOCA), assumptions (a-e) are made in addition to postulating the initiating event (see also FSAR Sections 6.3.2.5, 6.3.3 Safety Evaluation 9, and 15.6.5).
>
> a. The events are assumed not to result from a tornado, hurricane, flood, fire, loss of offsite power, or earthquake.
>
> b. Any one of the following occurs:
>
> 1. During the short term of an accident, a single failure of any active mechanical component. The short term is defined as less than 24 hours following an accident, or
>
> 2. During the short term of an accident, a single failure of any active or passive electrical component, or
>
> 3. A single failure of passive components associated with long-term cooling capability, assuming that a single active failure has not occurred during the short term. Long-term cooling applies to a time duration greater than 24 hours.
>
> c. No reactor coolant system transient is assumed, preceding the postulated reactor coolant system piping rupture.

d. No operator action is assumed to be taken by plant operators to correct problems during the first 10 minutes following the accident....

e. **All offsite power is simultaneously lost and is restored within 7 days** *[emphasis added]....*

f. For a LBLOCA, for additional safety no credit is taken for the functioning of non-seismic Category I components.

In the design and analysis performed for provision of protection of safety-related equipment from hazards and events (tornadoes, floods, missiles, pipe breaks, fires, and seismic events) which could reasonably be expected, the following assumptions were made (FSAR Section 3.6.1.1 describes the design bases relative to the evaluation of the effects of the pipe failure hazards discussed in Section 3.6.2.):

a. Should the event result in a turbine or reactor trip, loss of offsite power is assumed, and the plant will be placed in a hot standby condition.

b. If required by a Technical Specification limiting condition for operation or if the recovery from the event will cause the plant to be shut down for an extended period of time, the plant will be taken to a cold shutdown (CSD) condition.

c. Redundancy or diversity of systems and components is provided to enable continued operation at hot standby or to cool the reactor to a CSD condition. If required, it is assumed that temporary repairs can be made to circumvent damages resulting from the hazard. All available systems, including nonsafety-related systems and those systems requiring operator action, may be employed to mitigate the consequences of the hazard.

   In determining the availability of the systems required to mitigate the consequences of a hazard and those required to place the reactor in a safe condition, the direct consequences of the hazard are considered. The feasibility of carrying out operator actions are based on ample time and adequate access to the controls, motor control center, switchgear, etc., associated with the component required to accomplish the proposed action.

d. When the postulated hazard occurs and results in damage to one of two or more redundant or diverse trains, single failures of components in other trains (and associated supporting trains) are not assumed. The postulated hazard is precluded, by design, from affecting the opposite train or from resulting in a DBA. For the situation in which a hazard affects a safety-related component, the event and subsequent activities are governed by Technical Specification requirements in effect when that component is not functional.

e. When evaluating the effects of any earthquake, no other major hazard or event is assumed, and no seismic Category I equipment is assumed to fail as a result of the earthquake. Certain non-seismic Category I components are designed and constructed to ensure that their failure will not reduce the functioning of a safety-related component to an unacceptable safety level.

This criterion meets the intent of Regulatory Guide 1.29, Position C.2. Evaluation of component failure includes drop impact forces and secondary effects, such as spray and flooding from piping failure.

Paragraph 3.1.2e constitutes an assumption made, in addition to postulating the single failure that leads to the analyzed event. An alternate source of power is required to supply power to safety systems to mitigate the event while the preferred source of power is not available. As stated in Section 8.1.2, "Onsite Power System Description," of the Callaway FSAR (ADAMS Accession No. ML17076A375), the safety-related loads are powered from either the engineered safety feature (ESF) transformers or EDGs. Since EDGs serve as the standby power source in the case of a preferred power source (ESF transformer) outage, it can be inferred from the design assumption of paragraph 3.1.2e that the mission time, as defined above, of the EDGs is 7 days for a LOOP coincident with a DBA (i.e., loss-of-coolant accident, main steam line break, fuel handling accident, or steam generator tube rupture). Although the 7-day time is specified to apply to a LOCA, main steam line break, fuel handling accident, or steam generator tube rupture, there is no language addressing why this time should apply to only these particular DBAs. Additionally, although FSAR Section 3.1.2 is silent regarding offsite power restoration time with respect to external events, there is no language addressing why the 7-day time should not apply to these events.

The NRC staff reviewed the original Safety Evaluation Report (SER) dated October 1981, and the supplemental SERs (SSERs) (SSER 1, dated January 1982; SSER 2, dated June 1983; SSER 3, dated May 1984; and SSER 4, dated October 1984) in order to respond to the questions raised in this TIA. There were no definitive statements regarding EDG mission time or NRC acknowledgement and acceptance of the assumption of paragraph 3.1.2e of the FSAR identified during the examination of the original or SSERs.

In RG 1.9, Revision 4, which endorses IEEE Std 387-1995, the NRC has taken the position that 30 days is an acceptable assumption for the duration of an extended LOOP. Callaway, however, is committed to RG 1.9, Revision 1, which endorses IEEE Std 387-1977, for design and RG 1.9, Revision 3, which endorses IEEE Std 387-1984, for testing. The 30-day time period is not specified in RG 1.9, Revisions 1 or 3. Callaway has not agreed to follow RG 1.9, Revision 4, and it is not part of the licensing basis for Callaway.

Because the FSAR states the assumption that all offsite power is simultaneously lost and is restored within 7 days with respect to LOCA, main steam line break, fuel handling accident, or steam generator tube rupture and does not discuss a different assumption with respect to other DBAs or external events and because Callaway is not committed to RG 1.9, Revision 4, the NRC staff concludes that, for operability determinations, the Callaway EDG mission time is 7 days.

The NRC staff also reviewed the following elements of the plant's licensing basis; however, this did not change the staff's conclusion that, for operability determinations, the Callaway EDG mission time is 7 days.

- 10 CFR Part 50, Appendix A, GDC 17, requires an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. As such, either one of the power sources (assuming the other is not available) is required to operate for the duration of anticipated operational occurrences and postulated accidents to support operation of safety related equipment.

- Discussion of accident analysis in the FSAR Chapter 6, "Engineered Safety Systems," and Chapter 15, "Accident Analysis" (ADAMS Accession Nos. ML17053A212 and ML17065A090, respectively), which extends through the hot shutdown or cold shutdown modes of operation is generally considered stable conditions. Dose analysis in the FSAR is calculated through a period in which offsite releases of radioactive materials are postulated to occur (approximately 30 days). 10 CFR 50.46 ECCS acceptance criteria require long term core cooling to be maintained for extended duration. Therefore, a power source from EDGs may be required and maintained for extended duration to support long term core cooling.

- Postulated limiting accidents lead to the release of radioactive materials and the dose at the site boundary is used as an indicator to measure potential effects on public health and safety. To ensure that the dose is controlled and reduced in a rapid manner, the plant safety systems must be available and perform as designed to mitigate the consequences of the accident. Generally, accidents in the accident analysis are analyzed until hot shutdown or cold shutdown is reached (since that point is defined as stable).

- At Callaway, the systems relied upon to mitigate the consequences of an accident require electric power from onsite or offsite power sources. The onsite power source is designed to withstand natural phenomena such as seismic events and other abnormal operating occurrences and supply a reliable source of power to the safety equipment to mitigate the consequences of all postulated events and accidents. Callaway followed IEEE Std 308-1974 in its design. IEEE Std 308-1974 under paragraph 4.2, "Design Basis Event Effects," states, in part, "The Class 1E power systems shall be capable of performing their function when subjected to the effects of any design basis event." Earlier in the standard, the term "Design Basis Event" is defined as, "Postulated events specified by the safety analysis of the station used in the design to *establish the acceptable performance requirements* of the structures and systems" (emphasis added). In IEEE Std 308-1974, "Class 1E" is defined as, "The safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or otherwise are essential in preventing significant release of radioactive material to the environment."

2. How would the mission time differ absent the FSAR Section 3.1.2 assumption?

The NRC staff declines to answer this hypothetical question. The conclusions presented in this TIA response are not generic and do not directly apply to other licensees or sites; rather, they are specific to Callaway. Since the Callaway FSAR does in fact contain the Section 3.1.2 assumption, it is inappropriate to speculate about what the Callaway EDG mission time would be absent this assumption.

3. To what extent should the licensee have pre-established agreements, contracts, and procedures, to ensure that offsite power is restored in 7 days?

The Callaway licensing basis does not specify pre-established agreements, contracts, or procedures to ensure that offsite power is restored in 7 days. The Callaway licensing basis does not address these items and allows flexibility for the licensee to establish these so that power is restored within 7 days.

The regulation at 10 CFR 50.63, "Loss of all alternating current power," requires nuclear power plants to be able to withstand for a specified duration and recover from a station blackout. One of the factors considered for the plant-specific station blackout duration is the probable time needed to restore offsite power. Although station blackout is a beyond design basis event, the guidance applicable to the restoration of offsite power provides valuable insights to the answer to this question.

Concurrent with the development of the regulation at 10 CFR 50.63, and consistent with discussions with the NRC staff, NUMARC developed guidelines and procedures for assessing station blackout coping capability and duration for light water reactors – NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987 (ADAMS Accession No. ML102710587). The NRC staff reviewed the guidelines and analysis methods and concluded that NUMARC 87-00 provided guidance for conformance to 10 CFR 50.63.

NUMARC 87-00, Section 4.2.2, "AC Power Restoration," provides guidance for operations and load dispatcher personnel concerning the proper course of action for restoring AC power in a station blackout. Specifically, licensees should coordinate with the transmission system operator to have the following actions taken:

I. Load dispatchers should give the highest possible priority to restoring power to nuclear units. Procedures and training should consider several potential methods of transmitting power from blackstart capable units to the nuclear plant.
II. Should incoming transmission lines to a nuclear power plant be damaged, high priority should be assigned to repair and restoration activities to at least one line capable of feeding shutdown equipment.
III. Repair crews engaging in power restoration activities for nuclear units should be given high priority for manpower, equipment, and materials.

On February 1, 2006, the NRC staff issued Generic Letter (GL) 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006 (ADAMS Accession No. ML060180352). The GL requested licensees to determine if compliance is being maintained with NRC regulatory requirements governing electric power sources and associated personnel training. By letter dated August 1, 2006 (ADAMS Accession No. ML062200317), the

Callaway licensee provided a supplemental response to NRC question 7(a) of the GL, which states, "Briefly describe any agreement made with the TSO [transmission system operator] to identify local power sources that could be made available to resupply power to your plant following a LOOP event." Specifically, the supplemental response stated that the transmission system operator's operating manual provides proprietary guidance on how the Callaway switchyard would be supplied following a complete system blackout. This same guidance could also be used following a LOOP event. The supplemental response also stated:

> After the Callaway switchyard is energized, offsite power would be restored per plant emergency procedures. Emergency Operating Procedure Addendum 7, "Restoring Offsite Power," would be used by operators to restore offsite power to the plant. Operators would be directed to Addendum 7 by ECA-0.0, "Loss of All AC Power." Addendum 7 provides explicit direction on how to restore off site power to the plant once the switchyard has been energized.

Based on the above, the licensee should have agreements in place so that high priority will be given to necessary personnel, equipment, and materials to restore at least one transmission line to nuclear power units. The licensee should ensure load dispatchers give highest priority to this work.

4. <u>To what extent should non-safety, non-seismic Category 1 equipment (offsite power distribution and switchyard components) assumed for accident mitigation be included in the scope of equipment included in monitoring the effectiveness of maintenance as required by 10 CFR 50.65?</u>

The regulation at 10 CFR 50.65(b)(2)(iii) indicates that the scope of the monitoring program at Callaway shall include nonsafety-related SSCs whose failure could cause a reactor scram or actuation of a safety-related system.

Per Appendix 3A of the Callaway FSAR, Callaway follows Revision 2 of RG 1.160 as a method for complying with the provisions of 10 CFR 50.65. Revision 2 of RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," March 1997 (ADAMS Accession No. ML003761662), endorses NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," April 1996 (ADAMS Accession No. ML101020415). Section 8.2.1.5 of NUMARC 93-01 provides guidance on scoping equipment into the maintenance rule under 10 CFR 50.65(b)(2)(iii). In addition, Section 8.2.1 of NUMARC 93-01 states that, "The scope of the Maintenance Rule, as defined in 10 CFR 50.65 (b), is limited to SSCs that directly affect plant operations, regardless of what organization actually performs the maintenance activities. For example, electrical distribution equipment out to the first inter-tie with the offsite distribution system should be considered for comparison with [10 CFR] 50.65 (b), and thereafter, possible inclusion under the scope of the Maintenance Rule. Thus, equipment in the switchyard, regardless of its geographical location, is potentially within the scope of the Maintenance Rule."

Based on the above, SSCs at Callaway that directly affect plant operations, such as electrical distribution equipment out to the first inter-tie with the offsite distribution system, should be considered for scoping per 10 CFR 50.65, regardless of their geographical location

5. <u>What types of operator actions are allowed and are consistent with safety-related equipment mission time?</u>

There are no specific operator actions in Callaway's accident analysis for the EDGs. However, reasonable operator actions to correct minor problems consistent with the licensing basis and normal operations are permitted. The type of operator actions that are allowed and consistent with the safety-related equipment mission time are those that may be implemented as compensatory measures in accordance with the guidance in Section 07.03 of IMC 0326 and Section C.05 of Appendix C to IMC 0326. Briefly, compensatory measures that may be used to restore inoperable SSCs to an operable but degraded or nonconforming status are those that: have minimal impact on operation, are easily implemented, and are documented with a prompt operability determination.

The prompt operability determination must focus on the physical differences between the absence of operator action (or a former automatic action) and the ability of the manual action to accomplish the specific function or functions to support operability.

Some physical differences to be considered are the ability to recognize input signals for action, ready access to areas and recognition of set-points and design nuances that may complicate manual actions like automatic resets. The licensee should have written procedures of sufficient detail and personnel trained on the procedures before any manual action is substituted for the loss of an automatic action.

Additionally, the NRC expects that conditions calling for operator action as compensatory measures to restore SSC operability will be more quickly resolved since compensatory measures suggest a greater degree of degradation or non-conformance.

A more detailed discussion of the use of compensatory measures to restore potentially inoperable SSCs to an operable but degraded status can be found at Section 07.03 of IMC 0326. A more detailed discussion of the use of temporary manual actions in place of automatic action in support of operability is provided at Section C.05 of Appendix C to IMC 0326.

Based on the above, the NRC staff agrees that operator actions may be taken when plant conditions are stabilized to make minor repairs based on time and adequate access, including dose rates, to the areas and equipment required to accomplish the proposed action. Easily-implemented-procedure-controlled operator actions that operators are trained to perform and have a minimal impact on operation may be used.

5.0 <u>CONCLUSION</u>

Based on the evaluation described above, the NRC staff concludes the following:

1. The term "mission time" is not used in the Callaway licensing basis. However, because the FSAR states the assumption that all offsite power is simultaneously lost and is restored within 7 days with respect to loss-of-coolant accident, main steam line break, fuel handling accident, or steam generator tube rupture and does not discuss a different assumption with respect to other DBAs or external events and because Callaway is committed to RG 1.9, Revision 1, for design and RG 1.9, Revision 3, for testing, the NRC staff concludes that, for operability determinations, the Callaway EDG mission time is 7 days.

2. The NRC staff declines to answer this hypothetical question. The conclusions presented in this TIA response are not generic and do not directly apply to other licensees or sites; rather, they are specific to Callaway. Since the Callaway FSAR does in fact contain the Section 3.1.2 assumption, it is inappropriate to speculate about what the Callaway EDG mission time would be absent this assumption.

3. The licensee should have agreements in place so that high priority be given to necessary personnel, equipment, and materials to restore at least one transmission line to nuclear power units. The licensee should ensure load dispatchers give highest priority to this work.

4. SSCs at Callaway that directly affect plant operations, such as electrical distribution equipment out to the first inter-tie with the offsite distribution system, should be considered for scoping per 10 CFR 50.65, regardless of their geographical location.

5. The NRC staff agrees that operator actions may be taken when plant conditions are stabilized to make minor repairs based on time and adequate access, including dose rates, to the areas and equipment required to accomplish the proposed action. Easily-implemented-procedure-controlled operator actions that operators are trained to perform and have a minimal impact on operation may be used.

6.0 POTENTIAL OUTCOME PATHS

- Immediate implications: There are no safety concerns and therefore no foreseen immediate implications to this response.

- Generic Implications: Resolution of this issue does not warrant the issuance of a generic communication since the issue is specific to Callaway.

- Backfit Considerations: This answer does not constitute a backfit because it does not involve a new or different position from a previously applicable NRC staff position.

7.0 REFERENCES

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13. U.S. Nuclear Regulatory Commission, NUREG-0830, Supplement 2, "Safety Evaluation Report related to the operation of the Callaway Plant, Unit No. 1," June 1983.

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17. Callaway Plant, Unit 1 - Final Safety Analysis Report, Revision OL-22, Chapter 6, "Engineered Safety Features," dated April 4, 2017 (ADAMS Accession No. ML17053A212).

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19. Institute of Electrical and Electronics Engineers Standard 308-1974, "IEEE Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations," March 14, 1975.

20. Nuclear Management and Resources Council, Inc., NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," November 1987 (ADAMS Accession No. ML102710587).

21. U.S. Nuclear Regulatory Commission, Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006 (ADAMS Accession No. ML060180352).

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SUBJECT: RESPONSE TO TASK INTERFACE AGREEMENT 2014-10 RELATED TO THE REGULATORY POSITION ON EMERGENCY DIESEL GENERATOR MISSION TIME FOR OPERABILITY EVALUATIONS AT CALLAWAY PLANT, UNIT NO. 1 (CAC NO. MF5099, EPID L-2015-LRA-0001) DATED OCTOBER 19, 2018

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