UNION ELECTRIC COMPANY (dba AMEREN MISSOURI)

CALLAWAY PLANT

DOCKET NOS. 50-483 AND 72-1045

10 CFR 50.59 and 10 CFR 72.48 SUMMARY REPORT

Report Period: May 16, 2019 to June 15, 2021

EXECUTIVE SUMMARY

In accordance with 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2), a summary report has been prepared which provides summaries of the 10 CFR 50.59 and 10 CFR 72.48 evaluations of changes, tests, and experiments approved and implemented for activities at Callaway Plant.

This report covers all 10 CFR 50.59 evaluations for changes that were implemented from May 16, 2019 to June 15, 2021. During this period there were 11 changes implemented that required a 10 CFR 50.59 evaluation. For these changes, it was determined per 10 CFR 50.59(c)(1) that NRC approval was not required, and therefore, a summary of the 10 CFR 50.59 evaluation is hereby provided.

Additionally, this report covers all 10 CFR 72.48 evaluations for changes that were implemented from May 16, 2019 to June 15, 2021. During this period there was one change implemented that required a 10 CFR 72.48 evaluation. For this change, it was determined per 10 CFR 72.48(c)(1) that NRC approval was not required, and therefore, a summary of this 10 CFR 72.48 evaluation is hereby provided.

10 CFR 50.59 EVALUATIONS:

Evaluation Number:	**Activity:**
13-01	MP 03-1002, Main Feedwater Pump Turbine Control System Replacement, Rev. 2
17-01	MP 17-0006, ESW Water Hammer Mitigation
17-03	MP 15-0008, Open-Phase Condition Mitigation (with full trip capability enabled) (Ref.: NRC Bulletin 2012-01)
18-04	MP 08-0027, Main Turbine Generator Governor Controls Replacement
18-05	Allowance for manually bypassing UHS cooling tower automatic freeze protection
19-01	Reduction in Containment Passive Heat Sinks
19-02	MP 19-0103, Hot Leg Recirculation Valve Position Change and Mission Time Basis Documentation
20-02	MP 19-0092, Cycle 25 Core Reload with Framatome Lead Fuel Assemblies Included, Rev. 1
20-02	MP 19-0092, Cycle 25 Core Reload with Framatome Lead Fuel Assemblies Included, Rev. 2
20-05	RFR 200113, TMRE FSAR Update and Methodology Change
20-06	MP 19-0017, Inboard/Outboard Mechanical Seal Orifice Re-design for AFW Pumps

10 CFR 72.48 EVALUATIONS:

Evaluation Number:	**Activity:**
21-01	10 CFR 72.212 Evaluation Report for Callaway Plant

MP = Modification Package

10 CFR 50.59 Evaluation 13-01: MP 03-1002, Main Feedwater Pump Turbine Control System Replacement, Rev. 2

Activity Description:

Due to equipment obsolescence, the existing Westinghouse Steam Generator level control and Main Feed Pump Speed/Differential Pressure control system is being replaced with a digital feedwater control system. In addition to resolving obsolescence concerns, this new system removes multiple single-point vulnerabilities and provides additional functionality for improving plant control. The Main Feedwater Pump (MFP) controls on the main control board will be replaced by touch screen controls. The MFP Servo Positioner control equipment, Main Feedwater Regulating Valve (MFRV) positioners, MFRV Bypass Valve (MFRVBV) positioners, steam/feed flow recorders, and MFP local speed gauges will additionally be replaced. MFP speed, Steam Header Pressure (SHP), Feedwater Header Pressure (FHP), steam flow, feedwater flow, and narrow range SG level transmitter logic are to be modified as well. Redundant equipment will be installed for the MFRV positioners and FHP/SHP/MFP speed transmitters.

Summary of Evaluation:

This change has the potential to result in common mode software failures that are equivalent or not more than minimally impactful to the current FSAR described limiting failures for the FSAR Section 15.1.2 described “increase in feedwater” accident, FSAR Section 15.2.7 described “decrease in feedwater” accident, and FSAR Section 3B.4.2.3 described “pipe break in Area 5” hazard.

A new bounding FSAR Section 15.1.2 described “increase in feedwater” accident failure has been analyzed where a common mode software failure is postulated to open all of the valves while taking the pumps to full speed. In this Westinghouse reanalysis, however, this event's results feature mitigation via the high-high steam generator water level feedwater isolation followed by a low-low steam generator water level reactor trip, which is the same result as the previous, far more credible, bounding accident scenario. Therefore, the acceptance criteria for an ANS Condition II RCS heat-up event are still met. This evaluation has shown all of these conditions to be acceptable for implementation without the need for prior NRC approval.

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10 CFR 50.59 Evaluation 17-01: MP 17-0006, ESW Water Hammer Mitigation

Activity Description:

The configuration of the Essential Service Water (ESW) system is susceptible to pressure transients caused by a water column separation in the higher elevations that occurs when the water supply to the system is stopped and then subsequently restarted (as is the case during a Loss of Offsite Power event). When the water columns rejoin, a pressure wave is formed in the piping system as a result of the rapid change in fluid velocity. This pressure wave propagates throughout the piping system at a high speed in the fluid, creating unbalanced forces in piping segments separated between diameter or direction changes.

MP 17-0006 is a modification for mitigating the pressures and resultant forces from the water hammer brought about by the rejoining water columns. The modification adds pressurized accumulators at the Containment Air Coolers, Control Room Air Conditioning units, and the Component Cooling Water (CCW) heat exchangers. These accumulators inject non-condensable gas at low pressure to dampen the water hammer response. Water Hammer Mitigation Modification Support Analyses were developed to analyze the final mitigation design and to evaluate a set of sensitivity cases to ensure that the design will function over a range of initial operating set points.

The accumulators, piping, and valves to be interconnected to the ESW pressure boundary are classified as safety related. These components are designed to function during and following a design basis accident to mitigate the effects of the water hammer. The mitigation modification will restore piping and support design compliance with the 1974 Edition of ASME Section III with Summer 1975 Addenda, which is the site's licensing basis.

Summary of Evaluation:

FSAR SP Section 9.2.1.2 states that the ESW system is designed to remove heat from components important to mitigating the consequences of a LOCA or MSLB and to transfer the heat to the Ultimate Heat Sink (UHS). The ESW system operates in conjunction with the CCW system and other reactor auxiliary components and the UHS to provide a means to cool the reactor core and RCS to achieve and maintain a safe shutdown. The ESW system also provides emergency makeup to the Spent Fuel Pool and CCW systems, and is the backup water supply to the Auxiliary Feedwater system. In order to perform these functions 1) the pressure boundary of the ESW system must remain intact, and 2) the safety related heat exchangers, room coolers, and air conditioning units must remain capable of achieving the heat removal rates credited in the accident analysis.

The installation of air accumulators and their associated piping and valves will mitigate the column closure water hammer that occurs when the ESW system is in its standby alignment and the water supply to the system is stopped and then subsequently restarted (as is the case during a Loss of Offsite Power event). The air accumulators will return the ESW system to compliance with licensing basis piping and pipe support design requirements when placed in its standby alignment (such that it is capable of functioning on demand). Compliance with the licensing basis piping and support design ensures that the pressure boundary remains intact. Additionally, the use of air accumulators with a normally isolated fill line ensures that the accumulators cannot be overfilled and that the air mass injected during an accident is sufficiently small so that it cannot result in degraded heat removal in heat exchangers, room coolers, and air conditioning units cooled by the ESW system. The air accumulators are designed to prevent containment bypass and to limit ESW system inventory loss. Therefore, the ESW remains capable of achieving the heat removal rates credited in the accident analysis.

The proposed change meets applicable NRC requirements, including the design, material, and construction standards applicable to the SSCs being modified and installed. The installation of air accumulators in the ESW system does not result in a change that would cause any ESW system parameter to change. The ESW system design pressure, temperature, and flow rates are unaltered by this modification. The ESW system cannot initiate any accidents with or without the accumulators installed, and no new failure modes are introduced.

This change can be implemented without obtaining a license amendment because the proposed change was determined to not represent a more than minimal increase in the probability/likelihood of initiation of an accident or equipment malfunction from what is described in the FSAR.

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10 CFR 50.59 Evaluation 17-03: MP 15-0008, Open-Phase Condition Mitigation (with full trip capability enabled) (Ref.: NRC Bulletin 2012-01)

Activity Description:

The change addressed by this 10 CFR 50.59 Evaluation is for the safety-related portion of plant modification MP 15-0008 which was installed (but not yet enabled) to provide protection against the potential effects of an open-phase condition in the switchyard (high-side of the startup or either Safeguards Transformer), in accordance with Callaway's response to Bulletin 2012-01, "Design Vulnerability in Electric Power System." The change will be effected by activating (via installed switches) the Class 1E negative sequence voltage protective relays that were installed in the trip circuits for the off-site power source feeder breakers for each of the two 4.16-kV Class 1E buses.

Summary of Evaluation:

The proposed change is the enabling of the automatic protection capability of the open phase condition detection and protection system that was installed by MP 15-0008. This evaluation concludes that the automatic protection capability can be enabled without prior NRC approval. Spurious actuation of the automatic protection capabilities of the open phase detection and protection system could represent additional (spurious) transfers of the safety related electrical buses from their offsite sources to the onsite emergency diesel generators beyond what is currently described in the FSAR. However, initiation of the automatic protection function requires the two-out-of-two coincidence logic to be satisfied for either set of relays such that the spurious actuation of one train of protection would require the simultaneous failure of two independent safety related components, which is a beyond design basis consideration. Consequently, the proposed change was determined to not result in a more than minimal increase in the probability/likelihood of initiation of an accident sequence or equipment malfunction from what is described in the FSAR.

10 CFR 50.59 Evaluation 18-04: MP 08-0027, Main Turbine Generator Governor Controls Replacement

Activity Description:

This modification will replace the existing GE Mark II turbine control system, the turbine over-speed protection system, and the low load valve control system with the GE Mark VIe Digital Electro-Hydraulic Controls (DEHC) system. The existing control systems are analog and will be replaced with digital technology to eliminate single point vulnerabilities, address the limited support available for the Mark II system, address obsolescence, improve reliability, and reduce operator burden.

Summary of Evaluation:

Replacing the GE Mark II turbine control system, the turbine over-speed protection system, and the low load valve control system with the GE Mark VIe Digital Electro-Hydraulic Controls (DEHC) system does not require NRC approval prior to implementation. The Mark VIe control system can only impact the frequency of occurrence of the “increase/decrease in heat removal by the secondary system” design basis accidents (i.e., condition II events which cannot impact any condition III or IV events) through the false initiation of turbine trips, failure to initiate a turbine trip, or turbine runaway (over-speed). The frequency of inadvertent turbine trips is reduced through the use of triple redundant instrument channels and outputs for critical applications. At the same time, the application of Triple Modular Redundancy (TMR) technology minimizes the potential for failure to trip on demand. Even though operator interfaces and some controls are being automated, they are not being controlled or operated outside industry norms or outside the requirements specified in the FSAR. With the removal of the mechanical over-speed trip and the addition of the Trip Manifold Assembly (TMA) trip device, the system has been shown by calculation to be more reliable, thus affording the plant better missile generation protection. All potential failures of the new DEHC system are bounded within the existing accident analysis. Thus, the modification can be implemented without prior NRC approval.

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10 CFR 50.59 Evaluation 18-05: Allowance for manually bypassing UHS cooling tower automatic freeze protection

Activity Description:

The purpose of this change is to procedurally allow operation of the Essential Service Water (ESW)/ Ultimate Heat Sink (UHS) systems with the automatic freeze protection control circuits manually bypassed on one of the two trains (as there are times when such operation is needed on a temporary basis). Operation of the manual bypass switches is currently only allowed at above freezing conditions (>43.2 °F) in order to protect the associated UHS equipment during normal operations. However, if a DBA were to occur while in manual bypass, an operator can return the system to automatic control within 15 minutes to ensure freeze protection is maintained as described in the licensing basis.

Summary of Evaluation:

Manually bypassing one train of UHS freeze protection control circuits (for manually closing the associated UHS cooling tower bypass valve in order to drive water over the UHS cooling tower fill) does not require NRC approval prior to implementation. Use of the existing temperature limit for operation (> 43.2 °F) and a new operator action (for ensuring bypass is returned to automatic control within 15 minutes) will ensure the freeze protection function of the UHS cooling tower will continue to be met as currently described and analyzed in the FSAR.

The primary failure mode of concern for this change is an inability to open the valve to support freeze protection. However, this failure mode has already/always existed via the controls circuitry and physical valve operation and is thus not impacted by the use of the new hand-switch. In regard to any potential for valve malfunction due to the new/credited operator action, it has been shown that the there is a less than minimal increase in the likelihood of such a malfunction, as documented and deemed acceptable per the operator action review (CA2512a form) completed for this change. There is no impact to the consequences of an SSC malfunction. As this change cannot create or impact any initiator of an accident (existing or different type) and is not introducing new failure modes, there is no impact to the frequency or consequences of an accident by implementing this change.

10 CFR 50.59 Evaluation 19-01: Reduction in Containment Passive Heat Sinks

Activity Description:

The purpose of this change is to support Licensing Document Change Notice (LDCN) 21-0006 which in turn reflects a change (reduction) in the amount (i.e, assumed surface areas) of credited passive heat sinks in the containment per calculations ZZ-525 (Rev. 2) and ZZ-443 (Rev. 1, Add. 3). Due to the reduction in credited heat sinks in containment, the change involves small increases in the post-accident containment pressures/temperatures calculated for applicable design basis accidents such as a main steam line break (MSLB) or loss of coolant accident (LOCA).

The FSAR changes include revising tables in which the containment heat sink areas are listed (for a given thickness), as well as figures depicting containment temperature and pressure profiles following a MSLB or LCOA. In addition to FSAR changes, TS Bases Section 3.6.4 is being changed to raise the value given for the calculated peak containment pressure from a MSLB occurring at 2% power, from "46.2 psig" to "47.5 psig" in the Applicable Safety Analysis section.

Since bounding values for the surface areas of credited heat sinks in containment are intended to be presented in the FSAR, the proposed change is intended to provide for adequate/additional conservatism in the calculations so as to allow for occasional changes to the equipment inside containment without the need for frequent or repeated updating of the calculations.

Summary of Evaluation:

Ameren Missouri calculations ZZ-525, Rev. 2, and ZZ-443, Rev. 1, Add. 3 analyzed the effects of reduced, credited passive heat sinks in containment. The passive containment heat sinks are described in the FSAR (in terms of areas and thicknesses) and consist of walls, structures and equipment located in containment. They are credited to have an initial cooling effect in the event of a large break LOCA (LBLOCA), small break LOCA (SBLOCA), and a MSLB. These revised calculations demonstrate that the proposed change does not affect previously calculated values for peak post-accident containment pressure or temperature, except that there is an increase from 46.2 psig to 47.5 psig for the worst MSLB case. However, this peak value is still less than the peak calculated containment internal pressure for the design basis LOCA (Pa) of 48.1 psig. (Pa determines the containment leak rate limit assumed in the accident analysis, and therefore, dose calculations are unaffected by the proposed change.)

Because the new calculated peak pressure for worst case MSLB continues to be bounded by the Pa such that the margin between Pa and the containment design pressure is unchanged, and all requirements for equipment inside containment to be qualified to function in all accident scenarios are unchanged, the proposed change to Callaway's current licensing basis may be implemented without obtaining a License Amendment.

10 CFR 50.59 Evaluation 19-02: MP 19-0103, Hot Leg Recirculation Valve Position Change and Mission Time Basis Documentation

Activity Description:

Callaway desires to eliminate the active function for supporting hot leg recirculation from EJHV8840 while also providing a basis for a mission time for the Safety Injection (SI) pumps (that is less than 30 days). As a result, a new approach for hot leg recirculation has been developed and is being proposed to support long term core cooling (LTCC).

The LTCC analysis of record (AOR) requires ECCS recirculation flow to both the RCS hot legs (HL) and cold legs (CL) during the HL recirculation phase of ECCS operation. For HL recirculation, the residual heat removal (RHR) and SI pumps are aligned, leaving only the centrifugal charging pumps (CCPs) to provide the credited CL recirculation flow. The RHR pumps take suction from the containment sump while the CCP and SI pumps take suction from the RHR pumps. In this lineup, all three ECCS pump subsystems are required to operate for the duration of the LTCC mission time (of 30 days). The updated LTCC analysis documented under MP 19-0103 maintains the requirement for ECCS recirculation flow to both the RCS HL and CL during the HL recirculation phase of ECCS operation. However, the required ECCS flow is to be accomplished using an altered pump lineup, and as such, the described lineup will constitute a new/revised "procedure described in the FSAR." The SI pumps are still transferred to hot leg recirculation while the RHR pumps are left aligned for cold leg recirculation.

This simplified lineup removes the need for the CCPs to provide long term CL recirculation flow. However, if available, they can remain aligned for CL recirculation. The proposed alignment also ensures that in the long term, passive flow is available through both potentially idle SI pumps to the RCS hot legs. Following sufficient subcritical decay time, minimum flow requirement to the hot legs can continue to be met by either of the RHR pumps via passive flow through both idle SI pumps. These changes in LTCC ECCS pump lineup are largely below the level of detail given in the FSAR. However, several FSAR sections will be updated to increase the level of detail regarding the LTCC analysis and pump lineups.

Summary of Evaluation:

This 10 CFR 50.59 Evaluation concludes and documents that the proposed change for post-LOCA hot leg recirculation can be implemented without prior NRC approval. The proposed change eliminates the active function to support hot leg recirculation from EJHV8840 while also providing a basis for a reduced mission time for the SI pumps and CCPs. This change reduces the complexity of transferring the ECCS from the cold leg recirculation mode to the hot leg recirculation mode of operation following a DBA LOCA. Fewer valves are required to be manipulated, and the optimized valve lineup allows hot leg and cold leg recirculation flowrates to be met with fewer ECCS pumps in operation. The proposed change does affect the required operator actions for hot leg recirculation following a failure of the NB02 bus. The revised operator actions would involve a local manual action to mitigate this single failure. However, a CA2512a review was completed to evaluate the revised operator actions which include manually repositioning a valve approximately 13 hours into the LOCA sequence. Thirteen hours is more than adequate time to perform this action. The required operator actions are incorporated into existing procedures and the operator training program. In addition, the equipment being actuated is located in areas that are accessible to the operator post-LOCA and can be repositioned without use of any additional support personnel or equipment. The evaluation therefore demonstrates that the proposed local operator actions are acceptable and do not require prior NRC approval.

10 CFR 50.59 Evaluation 20-02: MP 19-0092, Cycle 25 Core Reload with Framatome Lead Fuel Assemblies Included, Rev. 1

Activity Description:

MP 19-0092, Rev. 001 provides the Core Reload Design for Cycle 25. For Cycle 25, the Callaway core will be refueled with Westinghouse 17×17 VANTAGE+ fuel product, along with four Framatome GAIA lead fuel assemblies (LFAs) that will be inserted into the Cycle 25 core.

The addition of the LFAs requires that the Reload Safety Evaluation performed by Westinghouse for a reload design be expanded to address the impact of the LFAs on the core design and resident fuel. It also requires evaluation of the LFAs operating in the Callaway Cycle 25 core, including confirmation that they meet the requirements of Technical Specification 4.2.1, "Fuel Assemblies," and meet all GAIA design requirements.

Summary of Evaluation:

Callaway has chosen to conservatively treat the LFAs as lead test assemblies (LTAs) as discussed in Technical Specification 4.2.1 and to apply the guidance of the NRC's letter to NEI, dated June 24, 2019, "Clarification of Regulatory Path for Lead Test Assemblies," identified as Reference 17 in the 10 CFR 50.59 evaluation. The proposed change (i.e., inclusion of the four LFAs in the Cycle 25 core) was thus determined to be a test or experiment not described in the FSAR (in accordance with the Reference 17 guidance), and a 10 CFR 50.59 evaluation is required. Additionally, the 50.59 screen performed for MP 19-0092 identified the following FSAR-described design functions to be adversely affected by the change.

1. The margin to grid crush for combined seismic-LOCA design loads is lower for the LFAs than is documented in the RSE and supplemental evaluations for the Westinghouse fuel.
2. The LFAs have a lower axial load limit for handling: 2.5 g for LFAs vs 4 g for co-resident fuel.

Though the LFAs have less margin than the resident fuel for the cases listed above, significant margin to failure remains such that the considered failures remain non-credible.

In addition, although the introduction of four LFAs is considered a test or experiment not described in the FSAR, the Technical Specification (TS) 4.2.1 criteria for allowing a limited number of LFAs/LTAs in non-limiting regions is met. Based on conformance with TS 4.2.1 and the guidance of the Reference 17 NRC letter, along with the result that "no" responses were provided to all of the applicable 10 CFR 50.59 evaluation questions/criteria, this change does not require prior NRC approval.

Note: Revision 1 of this evaluation allowed for operation only in MODEs 5 and 6.

10 CFR 50.59 Evaluation 20-02: MP 19-0092, Cycle 25 Core Reload with Framatome Lead Fuel Assemblies Included, Rev. 2

Activity Description:

MP 19-0092, Rev. 002 provides the Core Reload Design for Cycle 25. For Cycle 25, the Callaway core will be refueled with Westinghouse 17×17 VANTAGE+ fuel product, along with four Framatome GAIA lead fuel assemblies (LFAs) that will be inserted into the Cycle 25 core.

The addition of the LFAs requires that the Reload Safety Evaluation performed by Westinghouse for a reload design be expanded to address the impact of the LFAs on the core design and resident fuel. It also requires evaluation of the LFAs operating in the Callaway Cycle 25 core, including confirmation that they meet the requirements of Technical Specification 4.2.1, "Fuel Assemblies," and meet all GAIA design requirements.

Summary of Evaluation:

The use of LFAs alone necessitated the performance of a 10 CFR 50.59 evaluation. Callaway has chosen to conservatively treat the LFAs as lead test assemblies (LTAs) as discussed in Technical Specification 4.2.1 and to apply the guidance of the NRC's letter to NEI, dated June 24, 2019, "Clarification of Regulatory Path for Lead Test Assemblies," identified as Reference 17 in the 10 CFR 50.59 evaluation. The proposed change (i.e., inclusion of the four LFAs in the Cycle 25 core) was thus determined to be a test or experiment not described in the FSAR (in accordance with the Reference 17 guidance), and a 10 CFR 50.59 evaluation is required. Additionally, the 50.59 screen performed for MP 19-0092 identified the following FSAR-described design functions to be to be adversely affected by the change.

1. The margin to grid crush for combined seismic-LOCA design loads is lower for the LFAs than is documented in the RSE and supplemental evaluations for the Westinghouse fuel.
2. The LFAs have a lower axial load limit for handling: 2.5 g for LFAs vs 4 g for co-resident fuel.

Though the LFAs have less margin than the resident fuel for the cases listed above, significant margin to failure remains such that the considered failures remain non-credible.

In addition, although the introduction of four LFAs is considered a test or experiment not described in the FSAR, the Technical Specification (TS) 4.2.l criteria for allowing a limited number of LFAs/LTAs in non-limiting regions is met. Based on conformance with TS 4.2.1 and the guidance of the Reference 17 NRC letter, along with the result that "no" responses were provided to all of the applicable 10 CFR 50.59 evaluation questions/criteria , this change does not require prior NRC approval.

Note: Revision 2 of this evaluation allowed for operation in MODEs 1, 2, 3, and 4.

10 CFR 50.59 Evaluation 20-05: RFR 200113, TMRE FSAR Update and Methodology Change

Activity Description:

The proposed change is to incorporate the Tornado Missile Risk Evaluator (TMRE) methodology described in NEI 17-02, Revision 1B, into the plant's licensing basis via RFR 200113. This risk-informed methodology was developed for industry use by the Nuclear Energy Institute (NEI) to resolve tornado missile protection (TMP) nonconformances identified in response to Regulatory Issue Summary (RIS) 2015-06, "Tornado Missile Protection." The TMRE methodology was approved by the NRC for three pilot plants, i.e., Vogtle (Units 1 and 2), Shearon Harris, and Grand Gulf, with conditions and limitations described in the Safety Evaluations for those issued license amendments (ADAMS Accession Nos. ML18304A394, ML18347A385, and ML19123A014, respectively).

Summary of Evaluation:

This 50.59 Evaluation concludes that use of the TMRE methodology to evaluate and disposition tornado missile protection (TMP) nonconformances identified in response to RIS 2015-06 is not a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses. This conclusion is based on previous NRC approval for the use of TMRE per the license amendments and accompanying NRC Safety Evaluations for pilot plants Shearon Harris, Vogtle and Grand Gulf, as well as on guidance provided in the NRC's letter dated February 7, 2020 in which it was identified that TMRE may be considered a method of evaluation such that its use as an alternate method for evaluating TMP nonconformances can be evaluated under 10 CFR 50.59. The use of TMRE at Callaway is precisely for that purpose (i.e., for resolving the nonconformances identified at Callaway in response to RIS 2015-06) and is therefore compliant with the intended use approved by the NRC. Further, the Kinectrics TMRE report for Callaway (referenced in RFR 200113) documents that the conditions and limitations specified in the Safety Evaluations issued for the pilot plant license amendments were reviewed and have been applied or satisfied, as applicable, in the TMRE application to Callaway Plant.

10 CFR 50.59 Evaluation 20-06: MP 19-0017, Inboard/Outboard Mechanical Seal Orifice Re-design for AFW Pumps

Activity Description:

Plant modification MP 19-0017 calls for removal of the single orifice in the cooling water supply to the inboard and outboard mechanical seals on each of all three Auxiliary Feedwater pumps, PAL01A, PAL01B, and PAL02, and replacing it with an orifice in each individual seal line. In addition, for PAL02, the seal water piping will be replaced with tubing in order to minimize leakage (as described in RFR 180077). The tubing configuration for PAL02 requires performance of a pipe stress analysis in support of the intended design. The required pipe stress analysis is to be performed using the computer program “AutoPIPE,” which is new for Callaway and thus not one of the “methods of evaluation” (MOEs) identified in the Callaway FSAR for pipe stress analyses. The FSAR-described MOE that otherwise would have been used for this pipe stress analysis is PIPESTRESS, which is identified in Chapter 3 of the FSAR. The change thus involves replacing a method of evaluation described in the FSAR.

Summary of Evaluation:

Per this 10 CFR 50.59 evaluation, it was determined that use of the computer code AutoPIPE (Version 09.0601.11) in place of PIPESTRESS (PIPESTRESS2000) for the intended application does not require prior NRC approval. This conclusion is based on confirmation that AutoPIPE (Version 09.06.0 1.11) is an NRC-approved method for performing pipe stress analyses, based on an NRC Safety Evaluation (SE) issued in support of a license amendment approved for Three Mile Island Nuclear Station (ADAMS Accession No.ML15225A158), for which the noted version of AutoPIPE was used for the same type of application. No “terms or conditions” were identified by the NRC in connection with the use of AutoPIPE, per the noted SE. Since AutoPIPE is a methodology approved by the NRC for the intended application, the use of AutoPIPE (Version 09.06.01.11) and its incorporation into the plant’s licensing basis is not a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

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10 CFR 72.48 Evaluation 21-01: **10 CFR 72.212 Evaluation Report for Callaway Plant**

Activity Description:

Revision 5 to the Callaway 10 CFR 72.212 Report includes the following changes which are subject to 72.48 evaluation.

1. Update report with a change to the procedure for downloading to address a previously unreviewed system interaction (load hang-up) which adversely affects how SSC design functions described in the HI-STORM UMAX FSAR (UFSAR) are performed or controlled.
2. Incorporate Site Specific Thermal Evaluation RRTI 2990-002 Revisions 001 and 002 which provide site specific time limits for Mating Device drawer closure. This evaluation was performed using FLUENT version 18.1 instead of FLUENT version 6.3 which is the approved method of evaluation specified in the UFSAR chapter 4, section 4.4.1.
3. Incorporate Site Specific Thermal Evaluation RRTI 2990-011 Rev. 0 which provides a thermal evaluation to approve the use of temporary shielding. This evaluation was performed using FLUENT version 18.1 instead of FLUENT version 6.3 which is the approved method of evaluation specified in the UFSAR chapter 4, section 4.4.1.

Summary of Evaluation:

The procedure for downloading multi-purpose canister (MPCs) into their designated receptacles described in the UFSAR was revised to address a previously unreviewed system interaction (load hang-up). The current licensing basis considers an MPC drop event during download to be non-credible based on the use of handling equipment with redundant drop protection features. The 2018 SONGs event demonstrated that redundancy alone was not sufficient to preclude a drop. The proposed change adds and credits a load monitoring system (LMS) and credits procedure controls/manual action to be applied when site-specific LMS set points are reached, in order to preclude defeating the redundant drop protection features. Based on these added credited provisions, requiring changes to Callaway's procedure(s), it is concluded that there is no possibility for an accident of a different type (i.e., the drop of an MPC remains non-credible, consistent with the current licensing basis) and there is no increase in the frequency or likelihood of equipment malfunction associated with this change.

Review of the changes affecting the 10 CFR 72.48 report also identified two proposed changes as described above (changes 2 and 3) that involve revision of the same UFSAR-described evaluation methodology and thus require a 10 CFR 72.48 evaluation due to the methodology change. Site-specific thermal evaluations (to establish time limits with the mating device drawer closed and to approve the use of temporary shielding) require use of a later version of the FLUENT code than that described in the UFSAR. The change is evaluated as conservative or essentially the same and not a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.