Ameren Missouri's Response to Green NCV 05000483/2022010-03

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ENCLOSURE 1

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A. NCV 05000483/2022010-03

A detailed description and explanation of NCV 05000483/2022010-03 is provided in Section 71152B of the associated Inspection Report (dated October 12, 2022), pages 13-17. The description of this violation is taken directly from that section of the report and provided here for ease of reference.

1.0 Introduction

The inspectors identified a Green finding and associated non-cited violation of Title 10 CFR 50.55a(f), "Preservice and inservice testing requirements," paragraph (4), "Inservice testing standards requirement for operating plants," for the licensee's failure to perform the required inservice testing in accordance with the ASME OM Code for trains A and B residual heat removal heat exchanger air operated outlet and bypass valves. Specifically, the licensee failed to perform required inservice surveillance testing for the four pneumatically (air) operated valves as a result of incorrectly classifying them as passive valves.

1.1 Description

The inspectors performed a five-year review of residual heat removal (RHR) system operating experience and identified condition report (CR) 202102508. This condition report documented operating experience associated with the classification of the train A and B RHR heat exchanger air operated outlet valves (EJFCV0606/0607) and bypass valves (EJFCV0618/0619) in the inservice testing (IST) program, which implements the requirements of the ASME OM Code (2004 Edition through 2006 Addenda) as incorporated by reference in 10 CFR 50.55a. Specifically, the operating experience stated EJFCV 0606, 0607, 0618, and 0619 were incorrectly classified as passive valves in accordance with the ASME OM Code and did not receive required testing in accordance with Subsection ISTC, paragraph ISTC-5132. The licensee documented its evaluation of the operating experience in CR 202102508 and stated:

> These valves have a similar 'passive' classification. However, the station is licensed for hot standby (Mode 3) as their 'safe shutdown' with regards to scoping inservice testing components required to achieve and maintain 'safe shutdown.' Additionally, there was no credited Mode 4 loss-of-coolant accident (LOCA) analysis for the station that would support the condition described in the operating experience. Licensing Document Change Request (LDCR) 202100500 was in progress to clarify the working of the Technical Specification bases 3.5.3 with respect to no credited Mode 4 LOCA analysis applicable for the station. Therefore, the condition as described would not result in change of the mentioned valves from a 'passive' to an 'active' classification for the station's inservice testing program. No further action was required.

The inspectors reviewed the licensee's licensing and design basis, LDCR 202100500, TSTF-575-T, "Technical Specification Task Force Improved Standard Technical Specification Change Traveler," Revision 0, SNP(UE)-904, "[Emergency Core Cooling System (ECCS)] Performance during MODE 4 Operation," dated April 9, 1986, SNP(UE)-944, "LOCA in MODE 4 – Notification of NRC of Information," dated July 8, 1986, OTG-ZZ-00006, "Plant Cooldown Hot Standby to Cold Shutdown," Revision 82, and M-22EJ01, "Piping and Instrumentation Diagram Residual Heat Removal," Revision 64. The inspectors noted the following:

- The subject valves are repositioned from their accident position during MODE 4 operations. Specifically, RHR heat exchanger air operated outlet valves (EJFCV0606 and EJFCV0607) are normally in the open position to align the RHR system for ECCS operation. The RHR bypass valves (EJFCV0618 and EJFCV0619) are normally in the closed position to align the RHR system for ECCS operation. Procedure OTG-ZZ-00006 places the first train of RHR in shutdown cooling mode of operation when reactor coolant system (RCS) temperature is less than 350 degrees F and places the second train of RHR into shutdown cooling mode of operation when RCS temperature is less than 240 degrees F in accordance with procedures OTN-EJ-00001, Addendum 3, "Placing A RHR Train In Service for RCS Cooldown," Revision 25, and OTN-EJ-00001, Addendum 4, "Placing B RHR Train In Service for RCS Cooldown," Revision 24. This repositions valves EJHCV0606, EJHCV0607, EJFCV0618, and EJFCV0619 out of their accident and fail-safe positions. Additionally, neither these procedures nor other administrative controls restrict the amount of time the four subject valves may be configured out of their safety position. Therefore, the valves would have to be repositioned to their fail-safe position by manual operation (the valves are also capable of automatically repositioning on a loss of power or loss of air to the valves).

- The RHR system, including the subject valves have a function to mitigate the consequences of an accident. Specifically, UFSAR Chapter 15, "Accident Analysis," includes Section 15.6.5, "Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," and considers small break loss of coolant accidents. Table 15.0-8, "Operator Actions Required for Small and Large Break LOCAs," states:

  > The generic thermal-hydraulic analysis for the limiting MODE 4 SBLOCA in WCAP-12476, Revision 1, is supplemented by a plant-specific evaluation (Westinghouse letter SCP-10-31, dated May 11, 2010) which demonstrates that the minimum safeguards ECCS flow from one centrifugal charging pump (CCP) and one residual heat removal (RHR) pump can satisfy the MODE 4 small break LOCA ECCS flow requirements…"

Section 16.5.3, “ECCS Subsystems – MODE 4 Entry,” Subsection 16.5.3.1.2, “Bases,” provides additional similar discussion of WCAP-12476, Revision 1, and notes:

> That topical report also presents a generic bounding thermal-hydraulic analysis for the MODE 4 small break loss of coolant accident based on limiting representative plant parameters with the accumulators isolated. The assumed ECCS availability is based on one OPERABLE ECCS train consisting of a centrifugal charging subsystem and an RHR subsystem.

Additionally, SNP(UE)-904 and 944 are part of a vendor part 21 notification to the agency associated with shutdown LOCAs and further reinforce why RHR, including the subject valves, must be in the ECCS mode of operation to mitigate the consequences of an accident. These documents state, in part:

> Certain combinations of operator actions and equipment status were assumed in reaching the conclusion that ECCS performance would be acceptable in the event of a credible small break LOCA. Without these combinations or some other equivalent combination, the effectiveness of the ECCS cannot be assured for a small break LOCA.

Similarly, Technical Specification (TS) 3.5.3 and TS Bases 3.5.3, “Emergency Core Cooling Systems (ECCS) – Shutdown,” requires, “One ECCS train shall be operable” in MODE 4. This includes a note stating an RHR subsystem may be considered operable during alignment and operation for decay heat removal if the system is capable of being manually realigned to the ECCS mode of operation. This allowance for manual realignment is to ensure RHR can respond to the small break LOCA. TS Bases 3.5.3 also states, “With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers.”

- The ASME OM Code requires valves that are required to change position to accomplish a specific function in mitigating the consequences of an accident to be considered active valves. Specifically, ASME OM Code, Subsection ISTA, paragraph ISTA-2000, defines active valves as, “Valves that are required to change obturator position to accomplish a specific function in shutting down a reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident.” Table ISTC-3500-1, “Inservice Test Requirements,” specifies applicable requirements for active and passive valves with respect to exercise and stroke-time testing, leakage testing, position indication verification, and fail-safe testing. Additionally, ASME OM Code, Subsection ISTC, paragraph ISTC3510, “Exercising Test Frequency,” together with paragraph ISTC-5130, “Pneumatically Operated Valves,” subparagraph ISTC-5131(a) require, that active pneumatically operated active valves shall have their stroke times measured when exercised on a quarterly basis in accordance with ISTC-3500.

The inspectors reviewed the above information in consultation with the NRC Office of Nuclear Reactor Regulation and concluded the following:

- A LOCA or any design basis accident, such as a loss of offsite power, are credible events required to be mitigated in MODE 4 in accordance with the regulatory requirements and the licensee's licensing and design basis.

- RHR valves EJHCV0606, EJHCV0607, EJFCV0618, and EJFCV0619 would be repositioned manually to their accident position in the event of a LOCA in MODE 4. As a result, the valves should have been classified as active valves and tested in accordance with ASME OM Code, contrary to the licensee conclusions documented in CR 202102508.

- The subject valves have been classified as passive valves prior to and since 2004.

- The licensee should revisit the conclusions in LDCR 202100500, which state:

> The current TS 3.5.3 bases mischaracterized the licensing and design basis for the ECCS during shutdown (Mode 4). There are no design basis accidents or transient analyses that are initialized in the applicability of TS 3.5.3 and that credit ECCS train. There are no analyses that characterize the probability of occurrence of a design basis accident or transient in the TS 3.5.3 applicability. The ECCS in the TS 3.5.3 applicability does not satisfy the 10 CFR 50.36(c)(2)(ii) Criterion 3 description of "A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient."

B. Response to NCV 05000483/2022010-03

2.0 Response

Ameren Missouri respectfully disagrees with the identified violation and maintains that the ASME OM Code passive classification for the subject valves is consistent with Callaway's design and licensing basis requirements, which fully comply with the regulatory requirements.

The subject valves are maintained in their safety position (i.e., not repositioned) during operational modes that require them to support a plant shutdown to Callaway's licensed safe-shutdown condition. For Callaway, the licensed safe-shutdown condition is Hot Standby (Mode 3). The subject valves are also maintained in their safety position (i.e., not repositioned) during operational modes in which they are required to be capable of mitigating the consequences of Callaway's analyzed design basis accident (DBA) loss-of-coolant accident (LOCA). Callaway's DBA LOCA analysis of record assumes the plant is operating in Mode 1 (full power), as the bounding condition, but no Mode 4 LOCA analysis of record exists.

Ameren Missouri is concerned that the inspection report position constitutes a new interpretation of Callaway's design and licensing basis requirements for LOCA mitigation. This NRC position would require Ameren Missouri to change Callaway's licensing and design bases. Due to the absence of regulatory requirements or NRC approved guidance related to performance of a Mode 4 DBA LOCA analysis, it is unclear what analysis and NRC approval would be needed for Callaway to comply with this position.

In summary, Ameren Missouri respectfully disagrees that a violation of regulatory requirements occurred and maintains that Callaway fully complies with Title 10 CFR 50.55a(f), "Preservice and inservice testing requirements," paragraph (4), "Inservice testing standards requirement for operating plants," for the subject valves. Further, changing the valves' ASME OM Code classification to active is inconsistent with Callaway's historical and current licensing basis. In addition, there are currently no regulatory requirements for licensees to perform a LOCA analysis in Mode 4.

The relevant documents and regulations substantiating Ameren Missouri's position are discussed below.

<u>2.1 Mode 4 LOCA Analysis</u>

Enclosure 2 contains a Pressurized Water Reactor Owners Group (PWROG) letter, OG-22-187, that discusses the background regarding Mode 4 LOCA analysis for Westinghouse NSSS plants. The letter supports the position that the regulations do not require the analysis of a LOCA in any plant condition other than Mode 1 at full power.

The remainder of this section addresses regulations that do require a LOCA analysis, the standard review plan discussions on LOCA analysis, and Westinghouse Standard Technical Specification Bases discussions for Emergency Core Cooling System (ECCS) operability in various modes of operation. Westinghouse letters from the 1986 timeframe are discussed as well, which addressed NRC questions regarding Mode 4 operations. These letters led to the development of WCAP-12476, "Evaluation of LOCA During Mode 3 and Mode 4 Operation for Westinghouse NSSS," which was an industry attempt to address shutdown LOCA risk. Despite the NRC's acknowledgment that this WCAP was developed as part of a voluntary program to address risk, it was incorrectly incorporated into Callaway Technical Specification 3.5.3 Bases and FSAR Chapter 15. This error and inconsistency, along with similar situations across the industry, was recognized by the PWROG which led to the development of Technical Specification Task Force Traveler TSTF-575-T to resolve the issue. Ameren Missouri recognized the inconsistency created by incorporating WCAP-12476 into Callaway's TS 3.5.3 Bases and FSAR, and is in the process of correcting this issue via Licensing Document Change Request (LDCR) 202100500 to adopt TSTF-575-T.

Applicable Regulations

Appendix A to Part 50—*General Design Criteria for Nuclear Power Plants,* Criterion 35- *Emergency core cooling,* states:

"A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure."

General Design Criterion (GDC) 35 clearly requires a single failure of the ECCS to be assumed. Technical Specification (TS) 3.5.2, "ECCS- Operating," in NUREG-1431, Rev. 5 (Enclosure 2, Reference 1), requires two ECCS trains to be Operable during Modes 1, 2 and 3, which addresses the single failure requirement of GDC 35. However, TS 3.5.3, "ECCS Shutdown," in NUREG-1431, Rev. 5 only requires one train of ECCS to be Operable such that a single failure of the ECCS is not required to be assumed in Mode 4.

Based on the above, GDC 35 is not applicable to the ECCS requirements in Mode 4.

Appendix K to Part 50—*ECCS Evaluation Models,* states:

"I. Required and Acceptable Features of the Evaluation Models

A. *Sources of heat during the LOCA*. For the heat sources listed in paragraphs I.A.1 to 4 of this appendix it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications…"

Operation in Mode 4 when the reactor is subcritical is inconsistent with these requirements. Appendix K only requires the LOCA analysis to be performed at 102% of licensed power level, i.e., Mode 1, and does not require any analysis to be performed in any lower Modes of operation, i.e., Mode 4. (As explained later in this response, there are certain functions not required to be Operable in Mode 4 that are otherwise assumed or required to be available in order to fulfill DBA LOCA mitigation requirements in Mode 1.)

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Rev. 3

Section 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," states:

"III <u>REVIEW PROCEDURES</u>

For the review of the ECCS performance analysis, as presented in the applicant's SAR, the reviewer verifies the following:

5. The parameters and assumptions used for the calculations were conservatively chosen, including the following points:

A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2% to account for power measurement uncertainties, unless a lower level of uncertainty can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event."

Operation in Mode 4 when the reactor is subcritical is inconsistent with these requirements; therefore, the Standard Review Plan (SRP) only requires the LOCA analysis to be performed at 102% of the licensed power level, i.e., Mode 1, and does not require an analysis to be performed in any lower Modes of operation, i.e., Mode 4. Additionally, it cannot be assumed that the Mode 1 LOCA analysis is bounding of Mode 4 conditions, since the Safety Injection (SI) on Containment Pressure- High 1 and Pressurizer Pressure- Low ESFAS functions, as well as the SI accumulators, which are assumed to function in the LOCA analysis that is performed at 102% of the licensed power level, are not available in Mode 4.

The Bases for TS 3.5.2, "ECCS-Operating," in NUREG-1431, Rev. 5

The Applicable Safety Analyses (ASA) Section of the Bases states:

"Each ECCS subsystem is taken credit for in a large break LOCA event at full power (Refs. 3 and 4).'

In this regard, the Bases for TS 3.5.2 refer to the "FSAR, Chapter [15], Accident Analysis" via Reference 4 in the Reference Section of the Bases. Bases discussion refers to the Chapter 15 LOCA analysis that is performed at 100% power, i.e., in Mode 1.

In addition, the LCO Sections of the Bases states:

"In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train."

This obviously acknowledges compliance with 10 CFR 50, Appendix A, GDC 35, including its requirement for single-failure protection, which is Reference 1 in the Reference Section of the Bases.

In all, the Bases for TS 3.5.2 provide a basis for how the TS 3.5.2 ECCS requirements meet the applicable regulatory requirements and how they are consistent with the mitigation capability required for a DBA LOCA in Mode 1 (full power).

The Bases for TS 3.5.3, "ECCS-Shutdown," in NUREG-1431, Rev. 5

The NUREG-1431, Rev. 5 Bases for TS 3.5.3 are not as clear about the basis for the ECCS requirements in Mode 4 (compared to the Bases for TS 3.5.2), as there are some references and statements made in the TS 3.5.3 Bases that could be viewed as conflicting, as further shown below.

The ASA Section of the Bases for TS 3.5.3 states:

"The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section."

As discussed in the discussion above regarding the Bases for TS 3.5.2, the safety analysis referenced and discussed in the Bases for TS 3.5.2 is the Chapter 15 LOCA analysis performed at 100%, i.e., in Mode 1, which is not applicable to Mode 4. This Bases statement thus appears to conflict with that difference.

Notwithstanding the above-noted Bases statement, the ASA Section of the TS 3.5.3 Bases also states:

"Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii)."

There is no Mode 4 DBA that credits an ECCS train, as discussed above which is consistent with the fact that there is no Mode 4 LOCA analysis in Callaway's licensing basis. The LOCA DBA discussed in the Bases of TS 3.5.2 is performed at 100%, i.e., in Mode 1; therefore, an ECCS train in Mode 4 does not satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

The Reference Section of the Bases states:

"The applicable references from Bases 3.5.2 also apply."

As noted previously, however, two of the References in the Reference section of TS 3.5.2 are 10 CFR 50, Appendix A, GDC 35 and FSAR, Chapter [15], Accident Analysis, which are both not applicable to Mode 4.

The confusing or conflicting references in the STS Bases for STS 3.5.3 have prompted Ameren Missouri's interest in a "T-Traveler" prepared by the Technical Specification Task Force (TSTF), described later in this section.

Westinghouse Letter Numbers SNP (UE)-904 and SNP (UE)-944

These letters are referenced in Callaway procedure OTO-BB-00010, "Shutdown LOCA." These letters discuss a 1986 review of Mode 4 LOCA evaluations with respect to the operational status of plant equipment during Mode 4 operation, in response to NRC questions. SNP-(UE)-944 discusses that additional consideration of these issues would be performed by the Westinghouse Owners Group (WOG). These considerations culminated with generation of WCAP-12476 which was subsequently used as the analytical basis for ARG-2, "Shutdown LOCA" (which was translated into Callaway procedure OTO-BB-00010, as further explained below).

These Westinghouse letters precede the development of WCAP-12476 which was subsequently submitted for NRC approval and then withdrawn due to the NRC's position regarding treatment of shutdown risk and shutdown operation through voluntary licensee programs as documented in the previously discussed NRC letter dated December 28, 1999 (Enclosure 2, Reference 5).

WCAP-12476, Rev. 0, “Evaluation of LOCA During Mode 3 and 4 Operation for Westinghouse NSSS”

Topical Report WCAP-12476, Rev. 0 was submitted to the NRC for review and approval via WOG (now PWROG) letter OG-91-61, dated November 27, 1991 (Enclosure 2, Reference 2).

Letter OG-95-069, dated August 25, 1995 (Enclosure 2, Reference 3), referenced a conference call with the NRC staff (the Staff) that was held on August 16, 1995, stating that the Staff planned to address the topical report (WCAP) as part of the ongoing overall shutdown risk review program. The letter also discussed that the WOG was informed verbally by the Staff that preliminary findings from that program determined that Mode 3 and 4 LOCAs were not significant to shutdown risk. The letter further discussed the generic procedural response guidelines that contained operator actions to respond to these postulated events. The letter concluded that the review of Mode 3 and 4 LOCAs should be included in the NRC’s shutdown risk program and that the review of the WCAP should not be completed until after the completion of that program.

Letter OG-99-039, dated April 28, 1999 (Enclosure 2, Reference 4), withdrew the WCAP from NRC review. The letter discussed that in late 1997, the shutdown and low power issue was subsumed by proposed changes to the Maintenance Rule (MR), which would require licensees to assess the impact on safety functions prior to removing equipment from service, and that the MR would apply to shutdown conditions. The letter discussed that a generic abnormal response guideline, ARG-2, was prepared to respond to a LOCA in Mode 3 and 4 based on the WCAP. The letter also discussed that a formal review of the WCAP was not required and asked the NRC to confirm this.

In an NRC letter dated December 28, 1999 (Enclosure 2, Reference 5), the NRC discussed that they could not concur that the WCAP could be used as the basis to develop plant specific abnormal operating procedures and used to support various engineering evaluations related to shutdown operations because the WCAP was not reviewed by the staff. Therefore, no staff position was made regarding any conclusions in the WCAP.

An NRC letter dated June 23, 2000 (Enclosure 2, Reference 6), stated:

"…NRC review of WCAP-12476, 'Evaluation of LOCA During Mode 3 and Mode 4 For Westinghouse NSSS,' is not necessary because implementation of the issues discussed in the WCAP related to shutdown risk and shutdown operations, including loss of coolant in Modes 3 and 4, are being addressed by licensees through voluntary programs. It is the staff's position, consistent with the direction provided by the Commission in Staff Requirements Memorandum, 'Staff Requirements - SECY-97-168 - Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation,' dated December 11, 1997, that NRC oversight of licensee performance in the area of shutdown operations will be through the oversight (inspection) process. Accordingly, any issues that arise as a result of NRC oversight of shutdown operations will be addressed through the means provided by the oversight process."

Based on the WOG-NRC correspondence discussed above, and because WCAP-12476, Rev. 0 was not reviewed and approved by the NRC, the topical report should not be incorporated in the licensing basis as a Mode 4 LOCA Chapter 15 AOR. (This is the basis, in part, for LDCR 202100500 mentioned in Inspection Report 05000483/2022010.) Further, as stated by the Staff, Mode 3 and 4 LOCA are addressed through voluntary programs, (i.e., not by regulations.)

ARG-2, Shutdown LOCA

Revision 4 of the Emergency Response Guidelines (ERGs) includes updates to the Abnormal Response Guidelines (ARGs) including ARG-2, "Shutdown LOCA," which is implemented at Callaway as OTO-BB-00010, "Shutdown LOCA."

The analytical basis for ARG-2 is being updated from WCAP-12476 Rev. 1 to PWROG-19021-P which was developed by the PWROG under PA-ASC-1610 Rev. 0, "ARG-2, Shutdown LOCA Analysis Basis Improvements." The PWROG-19021-P analysis is intended to define and manage the shutdown risk of removing ECCS actuation or equipment from service while transitioning through Hot Standby (MODE 3) and Hot Shutdown (MODE 4).

The report serves as an analytical basis for operator actions and time-sensitive elements used in abnormal operating procedures. These changes to ARG-2 are based on ongoing overall shutdown risk and analysis insights gained subsequent to development of WCAP-12476 Rev. 1. These updates to the ARG-2 analytical basis show Callaway's continued efforts to manage shutdown risk.

The update to the ARG-2 analysis was not submitted for NRC review and approval, as it pertains to analysis of shutdown risk, is not required by regulation, and does not constitute a design basis Mode 4 LOCA analysis. Callaway is concerned that NRC's new position on Callaway's analyzed LOCA design basis accident would prevent or complicate adoption of this improved basis for ARG-2 by Callaway and the rest of the industry.

Technical Specifications Task Force Traveler TSTF-575-T, "Revise TS 3.5.3, ECCS-Shutdown, Bases"

The Summary Description in TSTF-575-T states:

"The Bases for Pressurized Water Reactor (PWR) (PWR) Standard Technical Specifications (STS) 3.5.3, 'ECCS – Shutdown,' are revised. The current Bases contain misleading information. The current Bases infer that there are analyses that credit the use of the Emergency Core Cooling System (ECCS) in shutdown modes to mitigate a loss of coolant accident (LOCA) or other design basis accidents (DBAs). However, that is inconsistent with the licensing and design basis of operating plants. The proposed change corrects the Bases to be consistent with the Final Safety Analysis Report."

The Bases for TS 3.5.3 should be revised to delete any discussions regarding a LOCA or other DBAs, and the statement in the Applicable Safety Analysis Section should be revised that an ECCS train satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) in Mode 4, not Criterion 3, as shown in TSTF-575-T.

The Bases changes to TS 3.5.3, in TSTF-575-T, can be made under the provisions of the TS Bases Control Program, because the changes revise incorrect information that is contained in the Bases.

Mode 4 LOCA Summary and Conclusions

The following are the conclusions and key points for Ameren Missouri's basis/position regarding LOCA/ECCS analyses in Mode 4 for Callaway.

- The NRC does not require a Mode 3 or Mode 4 LOCA analysis to be performed by the regulations, but instead, licensees are allowed to address these events with voluntary programs. WCAP-12476, Rev. 1 was not reviewed and approved by the NRC; therefore, it should not be incorporated in the licensing basis as a Mode 4

LOCA Chapter 15 AOR. As a result, Callaway is correcting this information as part of LDCR 202100500.

- Appendix K and the SRP only require the LOCA analysis to be performed at 102% of the licensed power level, i.e., Mode 1, and do not require an analysis to be performed in any lower Modes of operation. This is particularly true for Mode 4 since certain functions are not required to be Operable in Mode 4 (per the applicable TS requirements) that are otherwise assumed or required to be available for LOCA mitigation in Mode 1.

- As discussed above, there are no regulatory requirements that require a licensee to perform a LOCA analysis in Mode 4.

- GDC 35 is not applicable to the ECCS in Mode 4.

- The Bases for TS 3.5.3 should be revised to delete any discussions regarding a LOCA or other DBAs, and the statement in the Applicable Safety Analysis Section should be revised to indicate that an ECCS train satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) in Mode 4 and not Criterion 3, consistent with TSTF-575-T. The TS 3.5.3 Bases changes addressed in TSTF-575-T can be made under the provisions of the TS Bases Control Program, because the changes revise incorrect information that is contained in the Bases.

<u>2.2 Callaway's Safe Shutdown Licensing Basis</u>

Callaway's safe shutdown basis, which is hot standby, has not changed since initial operation. SNP-1722 letter (dated 3/15/78) documents a Westinghouse response to Bechtel concerning licensing questions about what equipment is required to achieve and maintain a safe plant condition. In this letter, Westinghouse states that the safe shutdown design basis for the SNUPPS units is hot shutdown (hot standby in the STS operational mode definitions) as it is for all other Westinghouse designed pressurized water reactors. Westinghouse also stated that hot shutdown is a safe and stable plant condition which is easily achieved and maintained. This position is reflected in the SNUPPS FSAR 5.4A.3 (from which section 5.4A.3 of the Callaway FSAR was derived).

Callaway's FSAR 5.4A.3, "Safe Shutdown Scenario," states:

"The safe shutdown licensing basis is hot standby and the safe shutdown design basis is cold shutdown. Should an event occur which would place the plant under a Limiting Condition of Operation, or if recovery from the event will cause the plant to be shut down for an extended period of time, the plant may be taken from a hot standby condition to a cold shutdown condition."

The distinction between the licensing basis and design basis of safe shutdown was added to the FSAR via FSARCN 95-059 in response to CR 199502105. Prior to this change and during initial licensing, the first sentence simply stated "the safe shutdown design basis is hot standby." The purpose of the FSAR change was to clarify the safe shutdown requirements in response to questions during a Quality Assurance Audit.

The capability to achieve cold shutdown is a design basis for the plant, thus ensuring further cool down (from hot standby) can be accomplished. Progression to cold shutdown is at the licensee's discretion but subject to commitments to RG 1.139 and Branch Technical Position (BTP) RSB 5-1, which require the plant's design capability to achieve cold shutdown with only safety related components.

Regarding Callaway's compliance with RG 1.139 and BTP RSB 5-1, the failure modes and effects of specific RHR components, in consideration of their capability to support or achieve cold shutdown, are listed in FSAR Table 5.4A-3. The descriptions for air-operated valves EJ-FCV-618/619 and EJ-HCV-606/607 (i.e., the valves identified in the NCV) have not changed between the SNUPPS FSAR and the current Callaway FSAR. The position, which is stated in FSAR Table 5.4A-3, is that these valves are not required for safety-grade cold shutdown operations.

FSAR SP 5.4A.3.2 describes the analyzed plant operation to achieve and maintain cold shutdown. For the final cooldown phase, the evaluation recognizes that the air-operated flow control valves may be non-functional, and includes discussion about available success paths to mitigate this condition, as stated below:

"The RHR pump is restarted to initiate the final cooldown phase. At this point, since the air-operated flow control valves may not be functional, administrative control is required to avoid excessive heat loads (and resulting excessive duty) on the component cooling water system. Two methods of achieving this control are: 1) only one RHR pump may be operated or two RHR pumps can be started/stopped, over an extended period of time, to limit the total heat load on the RHR heat exchangers, or 2) throttling of the CCW flow to the RHR heat exchanger can be accomplished. This will result in less flow, though at a higher temperature, back to the CCW heat exchangers."

2.3 IST Program

SNUPPS originally agreed to submit a pump and valve test program (later called IST program) to the NRC for review during a June 9-10, 1981, in-person meeting with the NRC's Mechanical Engineering Branch. This program was submitted on the docket for the Callaway Plant via SLNRC 81-089 on September 5, 1981. In that original program, the subject valves (EJFCV0618, EJFCV0619, EJHCV0606, and EJHCV0607) were classified as passive valves that do not require testing.

Following the submittal of SLNRC 81-089, SNUPPS representatives engaged in further discussions with the NRC's Mechanical Engineering Branch. In these discussions, the NRC staff indicated some changes to the program were required. The changes were submitted via SLNRC 81-107 on September 18, 1981. None of these changes were relevant to the subject valves of this violation.

On January 31, 1984, Callaway transmitted a copy of Revision 1 of the IST program to the NRC via ULNRC-0738. As noted in the transmittal, this revision of the program was reviewed during a Pump and Valve Operability Review Team visit the week of December 5, 1983, by both NRC staff and their consultants. The designation of the four subject RHR valves remained passive in this revision of the program.

As part of their review, the NRC issued a Request for Additional Information (RAI) to Callaway on July 10, 1986, titled "Request for Additional Information Related to the Callaway Station Pump and Valve Inservice Testing Program." In this RAI, the NRC asked the following regarding the subject valves:

> "4. Do valves EJ-FCV-606, 607, 618, and 619 have a required fail-safe position? If so, are they tested to verify this fail-safe actuation per IWV-3415? Are these valves required to change position in order to take the plan to the cold shutdown condition following an accident?"

No docketed response from Callaway to this RAI could be found. This is likely due to the details of the RAI response being discussed in person with the NRC. The cover letter of the July 10, 1986 RAI alluded to this by stating, "we have found from past experiences that acceptable responses can best be arrived at through a discussion between our staff, consultants, and your staff. Please notify us when you are ready for us to schedule a meeting to discuss your responses."

Callaway submitted an updated revision of the IST program to the NRC for approval on January 14, 1987 via ULNRC-1426. Review of the list of program in that letter did not identify any changes to the treatment of the four subject valves. This revision of the IST program was reviewed and approved by the NRC via letter dated October 26, 1987. The SER concluded, "Based on the review of your IST program revision and relief requests, the staff concluded that the IST program will provide reasonable assurance of the operational readiness of the pumps and valves covered by the IST program to perform their safety-related functions... "During the review of the licensee's revised inservice testing program, the staff has not identified any significant misinterpretations or omissions of Code requirements. Thus, the IST program update transmitted by letter dated July 28, 1987, is acceptable for implementation."

The question could be posed whether the NRC had a tacit understanding that both trains of RHR would not be aligned in cooldown lineup while in Mode 4. However, looking at Revision 0 of procedure OTG-ZZ-00006 (used to move the plant from Mode 3 to Mode 5) and Revision 0 of procedure OTN-EJ-00001 (used to place RHR trains in cooldown alignment), it is clear that since initial plant operation, both trains of RHR were to be placed in cooldown alignment while in Mode 4. In fact, in response to OE on RHR seal failures and concerns over the potential for flashing in the RHR suctions lines when swapping to SI line up, Callaway now places the second train of RHR in service at a lower temperature than original operation.

For the second ten-year interval of the Callaway IST program, the NRC contracted Brookhaven National Laboratory to develop a Technical Evaluation Report (TER) following review of the program. This TER is dated January 1995. As part of this report, the scope of the IST Program was reviewed for nine systems, including RHR. The report concluded, "in general, the scope of the Callaway IST program for selected systems appears complete; however, certain valves were identified that may be within the scope of IST. It is recommended that the following valves be reevaluated for inclusion in the IST Program if the valves are Code Class 1, 2, or 3 and the function of the valves is credited in the plant's safety analysis." For the RHR system, the only valves listed were relief valves EJ8856A/B, EJ8842, EJV156/157, and EJV084/85.

As demonstrated above, the NRC has reviewed the Callaway IST program for scope inclusion several times. Each time, the four subject RHR valves (EJFCV0618, EJFCV0619, EJHCV0606, and EJHCV0607) were classified as passive valves with no active testing requirements. This was found acceptable by NRC reviewers.

C. Conclusion

In conclusion, Ameren Missouri respectfully disagrees with NCV 050002022010-03 and maintains that the ASME OM Code passive classification for the subject valves is consistent with Callaway's design and licensing basis requirements, which fully comply with the regulatory requirements.

The subject valves are maintained in their safety position (i.e., not repositioned) during operational modes that require them to support a plant shutdown to Callaway's licensed safe-shutdown condition. For Callaway, the licensed safe-shutdown condition is Hot Standby (Mode 3). The subject valves are also maintained in their safety position (i.e., not repositioned) during operational modes in which they are required to be capable of mitigating the consequences of Callaway's analyzed LOCA design basis accident. Callaway's LOCA analysis of record assumes the plant is operating in Mode 1 (full power), as the bounding condition, but no Mode 4 LOCA analysis of record exists.

Additionally, Ameren Missouri is concerned that the inspectors' position on Callaway's analyzed LOCA design basis accident constitutes a new interpretation of Callaway's design and licensing basis requirements. This NRC position would require Ameren Missouri to change Callaway's licensing and design bases. Due to the absence of regulatory requirements or NRC approved guidance related to performance of a Mode 4 DBA LOCA analysis, it is unclear what analysis and NRC approval would be needed for Callaway to comply with this position.

In summary, Ameren Missouri respectfully disagrees that a violation of regulatory requirements occurred and maintains that Callaway fully complies with Title 10 CFR 50.55a(f), “Preservice and inservice testing requirements,” paragraph (4), “Inservice testing standards requirement for operating plants,” for the subject valves. Further, changing the valves' ASME OM Code classification to active is inconsistent with Callaway's historical and current licensing basis. In addition, there are currently no regulatory requirements for licensees to perform a LOCA analysis in Mode 4.