Ameren
MISSOURI

Callaway Plant

January 29, 2014

ULNRC-06071

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

10 CFR 50.55a

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RE: I3R-15, PROPOSED ALTERNATIVE REGARDING PRESSURE RETAINING BOUNDARY DURING SYSTEM LEAKAGE TEST (TAC NO. MF2921)

Reference 1: AmerenUE Letter ULNRC-06048, "10 CFR 50.55a Request: Proposed Alternative to ASME Section XI Code Requirements Regarding System Leakage Tests of Class 1 Piping and Components Isolated Between Normally Closed Valves" dated October 17, 2013

Reference 2: NRC Letter "Callaway Plant, Unit 1 – Request for Additional Information Re: I3R-15, Proposed Alternative Regarding Pressure Retaining Boundary During System Leakage Test (TAC No. MF2921)" dated December 24, 2013

By letter dated October 17, 2013 (Reference 1) and pursuant to 10 CFR 50.55a(a)(3)(ii), Ameren Missouri (Union Electric Company) requested approval of Relief Request I3R-15 regarding the scope of Class 1 pressure retaining piping and components required to be included in system leakage testing at a pressure corresponding to 100% power operation, as specified per Paragraph IWB-5222(b) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI. From its ongoing review of I3R-15, the NRC staff has transmitted a request for additional information (RAI) by letter dated December 24, 2013, (Reference 2) for which responses from Ameren Missouri are needed in order to support completion of the NRC's review.

Ameren Missouri's responses to the RAI questions are provided in the attachment to this letter.

This letter does not contain new commitments. If there are any questions, please contact me at 573-676-8719 or Tom Elwood at 314-225-1905.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

Executed on: 1 / 29 / 2014

S. A. M

S. A. Maglio,
Manager, Regulatory Affairs

JPK/nls

Enclosure

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING
RELIEF REQUEST I3R-15

1. **Please provide additional discussion as requested below for the basis for hardship for not meeting the ASME Code system leakage test requirements of IWB-5222(b) for the Group 1, 2, 3, 4, and 5 piping system identified in Section 1 of RFA 13R-15.**
   a. **Discuss potential personnel safety hazards that could be introduced if each group of piping is subjected to the system leakage test in accordance with IWB-5222(b). Examples include occupational hazards, risk for spills, contaminations, and exposure to unwanted excess radiations.**
   b. **Provide an estimate for person-rem (roentgen equivalent man) exposure with consideration of as low as reasonably achievable (ALARA).**
   c. **Discuss whether the leakage test in accordance with the IWB-5222 requirements could cause a reactor trip or violate any requirement(s) of the technical specifications (TS).**

Callaway Response

Group 1: The reactor coolant system drain lines (BB Drains) are all equipped with two manual isolation valves and only one line has a test connection for pressurizing with an external pressure source. Testing of the three drain lines not equipped with test connections would have to be performed while the plant is in Mode 3 or higher when the reactor coolant system is at the pressure associated with 100% reactor power (2235 pounds per square inch gauge(psig)) at which time the temperature can be up to 558°F. An operator would have to be posted at the inboard isolation valve to both open the valve and act to close the valve should any downstream leakage occur. Additional personnel would have to be posted at the downstream piping to monitor for leakage should the outboard isolation valve leaked by. Stationing personnel to operate the manual valves and monitor the lines poses a physical safety hazard given the piping is at a pressure of 2235 psig and a temperature of 558°F.

Radiation exposure for performing this test on the three lines not equipped with a test connection is estimated at 100 mRem based on general area dose rates in the respective areas.

Additionally, opening the isolation valves would violate the double isolation requirement of 10 CFR 50.55a(c)(II) because these valves are not capable of automatic actuation.

The drain line that is equipped with a test connection could be tested with an external pressure source while the plant is in a lower mode and therefore reduce personnel exposure to the temperature and pressure of the main reactor coolant system piping. The external pressure source, a hydro pump, would still expose personnel to physical hazard from the temporary high pressure connections. Radiation exposure for set up, performance and removal of the test equipment is estimated to be 80 mRem.

Groups 2, 4, and 5, EM Cold Leg Injection, EJ Suction, and BG Auxiliary Spray and Normal Charging respectively, all have check valves for the inboard isolation. All the lines have test connections to allow pressurization with an external pressure source. To achieve the pressure required by IWB-5222(b), the test could be performed with use of a hydro pump while the plant is in Mode 3 at a pressure of 2235 psig. Additional personnel radiation exposure would not be significant as this could be performed as part of the Pressure Isolation Valve (PIV) testing performed at the end of each refueling outage. The PIV testing is normally performed at pressures from 300 to 1500 psig, so testing at a pressure of 2235 psig would expose personnel to additional physical hazard created by the high pressure temporary connections and hoses. A borated water source would have to replace the normal non-borated reactor make up water normally used for PIV testing. The only borated water sources readily available at these locations are from system piping which would cause concerns with contamination should any of the temporary fittings leak. The plant piping would also be exposed to additional degradation because the hydro pump would have to meet or exceed system pressure which would cause flow past the inboard check valves and the cold water being pushed into the hot reactor coolant system would cause thermal fatigue to the adjacent piping and welds.

Alternatively, the test could be performed while the plant is in a lower mode by disassembling and gagging closed the inboard check valves and then pressing up the line sections with a hydro pump. In addition to the physical hazard by potential failure of the temporary test connections for the hydro pump, disassembly of these valves poses significant personnel contamination concerns because the valve internals are highly contaminated. The radiation dose estimate associated with gagging the valves closed, including activities such as scaffold, insulation removal, and valve disassembly and reassembly, is estimated to be 5600 mRem based on general area dose rates and assuming the valve maintenance activities proceed without complications.

Attempting to block open or remove the internals of the check valves in order to use reactor coolant system pressure to pressurize the lines while the plant is in Mode 3 would cause non-compliance with TS 3.4.14, RCS Pressure Isolation Valve (PIV) Leakage.

Group 3: The EJ Suction piping sections are isolated by inboard and outboard motor-operated valves. Each section contains test connections that would allow pressurization from an external source. A hydro pump can be connected to each section and used to pressurize to 2235 psig. The personnel radiation exposure for this test should be less that 10 mRem for equipment setup and pressurization of the line segments. Hazards associated with this test would be exposure of personnel to the temporary high pressure connections for the hydro pump and personnel and equipment contamination concerns with venting and draining the EJ system piping. The EJ system is not rated to 2235 psig so would require venting or draining to ensure the system components would not be damaged should the isolation valves leak by.

Performing the test on this section by opening the inboard isolation valve to pressurize this section while the plant is in Mode 3 would require defeating the interlocks used to ensure these valves remain closed when reactor coolant system pressure is above 360 psig. Defeating these interlocks would cause non-compliance with TS 3.4.14. Opening the inboard isolation valves while in Mode 3 would also violate the double isolation requirement of 10 CFR 50.55a(c)(II).

2. **Discuss whether alternative options, such as the use of external pressure source, and temporary or permanent modifications, such as installing temporary or permanent piping to facilitate performance of the required system leakage test in accordance with IWB-5222, have been considered and determined to be not practical options.**

   Callaway Response

   All pipe sections listed in I3R-15, except for three of the BB Drains, have test connections available that would allow use of an external pressure source. For the BB Drains without test connections, adding test connections is not desirable. The additional piping, welds and valves would add to the complexity of the segments and therefore increase the probability of failure. Personnel radiation exposure would be increased both from the initial modification installation and also long term because the additional piping would make flushing radioactive contaminants from the lines less effective thereby increasing long term general area dose rates.

3. **For justification that the structural integrity or leak tightness of each group of piping identified in RFA 13R-15 will be reasonably ensured, without the required extension in pressure retaining boundary during system leakage test, discuss whether there has been any industry or plant-specific operating experience regarding potential degradation of the welded connections in the piping and components identified in RFA 13R-15 due to known degradation mechanisms that would lead to leakage.**

   Callaway Response

   In addition to the Callaway operating experience listed in I3R-15, searches of industry operating experience show occurrences of outside diameter stress corrosion cracking (ODSCC) as well as vibration and thermal fatigue in piping adjacent to the subject lines, but no failures have been identified within the boundaries of the subject lines. Callaway is following the guidelines of Electric Power Research Institute (EPRI) Materials Reliability Program (MRP)-146 to address thermal fatigue in reactor coolant system branch lines, EPRI MRP-192 for thermal fatigue in residual heat removal system mixing tees, and has identified and examined lines for vibration fatigue, but these issues are all outside the scope of the subject piping. ODSCC is applicable to the subject lines and Callaway is taking action to address this. Callaway is following the inspection guidelines in NRC Information Notice 2011-04, Contaminants and Stagnant Conditions Affecting Stress Corrosion Cracking in Stainless Steel Piping in Pressurized Water Reactors, and is also following the guidelines in the Pressurized Water Reactor Owners Group (PWROG) letter PA-MSC-0474, Outside Diameter Initiated Stress Corrosion Cracking Revised Final White Paper. Callaway is also going beyond this guidance and performing focused surface examination and ultrasonic testing on selected locations and has worked with EPRI to develop non-destructive examination techniques to detect ODSCC on stainless steel under pipe clamps.