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Donald F. Schnell
Senior Vice President
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March 11, 1992

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Gentlemen: ULNRC-2587

DOCKET NUMBER 50-483
CALLAWAY PLANT
REVISION TO TECHNICAL SPECIFICATION
3/4.4.4. AND 3.4.9.3

Reference: ULNRC-2527, dated December 4, 1991

The referenced letter transmitted Union Electric Company's amendment application to address the recommendations of Generic Letter 90-06.

Per our discussions with staff we are submitting additional information to support the significant hazards evaluations previously transmitted. The attached Significant Hazards Evaluation should be used to replace the previously submitted evaluation.

If there are any questions concerning this matter, please contact me.

Very truly yours,

Donald F. Schnell

Donald F. Schnell

JMC/plh
Attachment

A019
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STATE OF MISSOURI)
)   S S
CITY OF ST. LOUIS)

Donald F. Schnell, of lawful age, being first duly sworn upon oath says that he is Senior Vice President-Nuclear and an officer of Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Donald F. Schnell
Donald F. Schnell
Senior Vice President
Nuclear

SUBSCRIBED and sworn to before me this 11th day of March, 1992.

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BARBARA J. PFAFF
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MY COMMISSION EXPIRES APRIL 22, 1993
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SIGNIFICANT HAZARDS EVALUATION

This amendment application requests a change to Technical Specifications (T/S) 3/4.4.4 on Relief Valves and its associated Bases and T/S 3.4.9.3 on Overpressure Protection System. These specifications are being revised to incorporate certain NRC staff positions resulting from the resolution of Generic Issue 70 and 94 as presented in Generic Letter (GL) 90-06. The proposed changes to T/S 3/4.4.4 are as follows:

1. ACTION statement a. is being revised to include the requirement to maintain power to closed block valve(s) because removal of power would render the block valve(s) inoperable and the requirements of ACTION d. would apply. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage. Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.

2. ACTION statements a., b., and c. are being changed to terminate the forced shutdown requirements with the plant being in HOT SHUTDOWN rather than COLD SHUTDOWN because the APPLICABILITY requirements of the LCO do not extend past the HOT STANDBY mode.

3. ACTION statement d. is being changed to establish remedial measures that are consistent with the function of the block valves. The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve(s) cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The modified ACTION statement does not specify closure of the block valves because such action would not likely be possible when the block valve is inoperable. Likewise, it does not specify either the closure of the PORV, because it would not likely be open, or the removal of power from the PORV. When the block valve is inoperable, placing the PORV in manual control is sufficient to preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve.

The proposed changes to Technical Specification 3/4.4.4 and its Bases do not involve a significant hazards consideration because operation of Callaway Plant with these changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. No credit

is taken for operation of the PORVs in the FSAR Chapter 15 accident analyses if their operation mitigates the result of the accident. Turbine trips are evaluated in FSAR Section 15.2.3 with and without the pressurizer PORVs. The loss of offsite AC power and loss of normal feedwater analyses (FSAR Sections 15.2.6 and 15.2.7) assume the PORVs are operable only because their operation maximizes the transient pressurizer water volume caused by condensation of steam that would have been relieved through the safety valves. The proposed changes to Technical Specification 3/4.4.4 Action a. requires that with the block valve(s) closed, power be maintained to the valve(s) so they can be readily opened from the control room. This change would decrease the amount of time needed to initiate feed and bleed capabilities in the event an alternative measure to remove decay heat from the reactor core is necessary. The proposed change to T/S 3/4.4.4 Action d. is a clarification for a potential situation where an automatic signal to the PORVs is inoperable but the PORV is mechanically functional. Since the PORV is still mechanically functional, it would enhance safe operation to not close and remove power from the block valve, and allow the PORV to remain in a condition where it could easily be manually opened from the control room if required. This clarification is consistent with the operability requirements for the PORVs in Modes 1, 2 and 3. Therefore, the proposed changes to Technical Specification 3/4.4.4, and its associated Bases are intended to increase the reliability and availability of the PORVs, and do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated. There is no new type of accident or malfunction being created and the method and manner of plant operation remains unchanged. No change in testing methodology is being proposed, and the equipment is not being operated in a new or different manner. Changes incorporate the staff positions delineated in GL 90-06.

3. Involve a significant reduction in margin of safety. There are no plant design changes involved and no changes are being made to the safety limits or safety system settings that would adversely impact plant safety. The proposed changes to Technical Specification 3/4.4.4 increase the availability and reliability of the power-operated relief valves (PORVs) and block valves to perform their intended function. The changes do not reduce any technical specification margin of safety.

The proposed changes to T/S 3.4.9.3 are as follows below:

1. The LCO statement is being modified to require that at least two overpressure protection devices must be operable. That is, two PORVs or two Residual Heat Removal (RHR) Suction relief valves or one PORV and one RHR suction relief valve

must be operable when cold overpressure protection is required. At Callaway, the operability of two PORVs, of two RHR suction relief valves, of one RHR suction relief valve and one PORV, or a reactor coolant system vent opening of at least 2 square inches ensures that the RCS will be protected as required by 10CFR50, Appendix G.

2. ACTION statement a. is revised to clarify that it is only applicable in Modes 3 or 4. At Callaway, the overpressure protection system is required to be operable in Mode 3 below 368°F and in Modes 4, 5, and 6. This revision to the ACTION statement makes it only applicable in Modes 3 or 4 when only one pressure relief device is operable. This is consistent with GL 90-06 Attachment B-1.

3. ACTION statement b. is added to reduce the allowed outage time (AOT) for one of the two required PORVs or RHR suction relief valves to 24 hours in Modes 5 or 6. That is, 2 PORVs or 2 RHR Suction relief valves <u>or</u> 1 PORV and 1 RHR suction relief valve must be operable when cold overpressure protection is required. The NRC has considered the conditions under which a low-temperature overpressure transient is most likely to occur. While low-temperature overpressure protection is required for all shutdown modes, the most vulnerable period of time was found to be Mode 5 with the reactor coolant temperature less than or equal to 200°F, especially when water solid, based on the detailed evaluation of operating reactor experiences performed in support of GI 94. The staff concluded that the low-temperature overpressure protection system performs a safety-related function and inoperable overpressure protection equipment should be restored to an operable status in a shorter period of time. The current 7-day AOT is considered by the NRC to be too long under certain conditions. The NRC has concluded that the AOT should be reduced to 24 hours when operating in Modes 5 or 6 when the potential for an overpressure transient is highest.

The proposed changes to Technical Specification 3.4.9.3 do not involve a significant hazards consideration because operation of Callaway Plant with these changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. There is no change being proposed in the control designed to limit the occurrence of an overpressure transient. The proposed changes to Technical Specification 3.4.9.3 only serve to limit the amount of time the plant is vulnerable to a potentially damaging overpressure transient with limited overpressure protection available. Therefore, the proposed changes to Technical Specification 3.4.9.3 increases flexibility and availability of the low-temperature overpressure protection system with a resultant increase in

the level of plant safety and do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated. There is no new type of accident or malfunction being created and the method and manner of plant operation remains unchanged. There are no changes being proposed to the level of surveillance required to demonstrate compliance with the LCO. The installed overpressure mitigation devices will continue to be operated and tested in a manner consistent with their design and installation. The proposed changes are intended to enhance the level of overpressure protection during periods of vulnerability. Changes incorporate the staff positions delineated in GL 90-06.

3. Involve a significant reduction in a margin of safety. There are no plant design changes involved and no changes are being made to the safety limits or safety system settings that would adversely impact plant safety. The proposed changes to Technical Specification 3.4.9.3 increases the flexibility and availability of the overpressure protection system to mitigate a low-temperature overpressurization event. The changes do not reduce any technical specifications margin of safety.

Based on the above discussions, it has been determined that the requested Technical Specification revisions do not involve a significant increase in the probability or consequences of an accident or other adverse condition over previous evaluations; or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.