Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

This change reduces the power level at which the reactor may be operated with one or more main steam safety valves (MSSVs) inoperable, to ensure that the secondary system is not overpressurized during the most severe pressurization transient of the secondary side. Additionally, this change will combine the TS action statements for 3- and 4-loop operation with one or more MSSVs inoperable, revise the mode requirements and times of Action Statement 3.7.1.1.a, and correct a reference in the bases section to Table 3.7–1. Reduction of the high neutron flux (HNF) trip setpoint will continue to be used as the means to ensure that the required reactor power level reductions are met. Mode 3 will be limited to application when the reactor trip breakers (RTB) are closed. Lack of NIS trip setpoint adjustments with the RTB open has no effect on the accident analysis. There is no change to the function of the MSSVs by the proposed change. This change will not alter any accident analysis assumptions or results for SQN. The proposed change will reduce the amount of relief capacity required to mitigate the consequences of the transient by reducing the total amount of energy in the primary system. Therefore, this change will not increase the probability of an accident.

This change is consistent with current SQN accident analysis assumptions for the MSSVs and does not change the containment response for any design basis event. Therefore, no change in the mitigation of an accident will result from this proposed change and no change will occur in the consequences of any accident currently analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

Inadvertent opening of a MSSV is currently analyzed as an initiating event for accidental depressurization of the main steam system. The proposed change does not alter the valves or any other plant component. The valves will continue to perform as analyzed in current accident analyses. The proposed change will not create the possibility for any new or different kind of accident.

By retaining the use of the HNF trip setpoint reduction, no change is being proposed in the methodology used to ensure that power reductions are carried out; therefore, this will not create the possibility of placing the plant into any new unanalyzed condition. Not adjusting the Nuclear Instrumentation System trip setpoint with the RTBs open will not create an accident. The existing accident analysis is still bounding.

Combining the separate action statements for 3- and 4-loop operation into a single action does not create the possibility for a new or different kind of accident. Operation with 4 loops will continue to be required in Modes 1 and 2 by TS 3.4.1.1.

Operation with less than 4 loops will continue to be governed by TS 3.4.1.2 in Mode 3 and TS 3.4.1.3 in Mode 4. This change will not place the plant in a configuration not currently bounded by existing accident analysis.

Revising the mode requirements and their associated times, consistent with the requirements in NUREG–1431, will continue to ensure that if the unit is unable to comply with the limiting condition for operation, the unit will begin an orderly shutdown until a mode is reached where the specification is not applicable.

3. Involve a significant reduction in a margin of safety.

The proposed change reduces the total energy of the reactor coolant system that will ensure the ability of the MSSVs to perform their intended function as assumed in current accident analyses. This change has been evaluated on a generic basis for Westinghouse Electric Corporation designed 4-loop nuclear steam supply systems. SQN plant specific features have been evaluated including power limit calculations and the interaction of the reactor protection system trip time delay and the anticipated transient without scram mitigating system actuation circuitry. Correcting this nonconservatism restores the margin of safety to what was originally envisioned. Therefore, the margin of safety assumed in the accident analysis is not reduced by this change.

Combining the separate action statements for 3- and 4-loop operation into a single action has no effect on the margin of safety for 4-loop operation with one or more MSSVs inoperable. Under the revised TS, 3-loop operation with one or more MSSVs inoperable would only be allowed in Mode 3, and 4-loop operation will be required in Modes 1 and 2 in accordance with current TSs 3.4.1.1 and 3.4.1.2.

Revising the mode requirements and their associated times, consistent with the requirements in NUREG–1431, will not reduce the safety margin since the new requirements will continue to place the unit in a mode where the TS is no longer applicable. The new completion times for mode changes are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

The margin of safety is unaffected by modifying the limits of Mode 3 applicability to require the RTBs to be closed as the intended safety function is already completed when they are open.

The NRC has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

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NRC Project Director: Frederick J. Hebdon.

Union Electric Company, Docket No. 50–483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: December 9, 1994, and January 27, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.6.1.2.a and its associated Bases. The changes would defer the next scheduled containment integrated leak rate test (CILRT) for one outage, from Refuel 7 (March 1995) to Refuel 8 (scheduled for September 1996).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report.

The Callaway CILRT history provides substantial justification for the proposed test schedule. Three CILRTs have been performed over a seven year period with successful results. The tests indicate that Callaway has a low leakage containment. There are no structural mechanisms which would adversely affect the structural capability of the containment and that would be a factor in extending the CILRT schedule by one refueling outage.

A risk impact assessment was performed, and a determination was made that there is insignificant risk impact as a result of changing the CILRT schedule. Containment leak rate testing is not an initiator of any accident, the proposed interval extension does not affect reactor operations or the accident analysis, and has no radiological consequences. There will be no changes to 10 CFR 100 dose limits or the control room dose limits. Extending the test interval will not, by itself, increase the probability of a malfunction of equipment important to safety. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated in the Safety Analysis Report.

There are no design changes being made that would create a new type of accident or malfunction. The proposed change will not alter the plant or the manner in which it is operated. The change revises the schedule for performing the periodic CILRT. The purpose of the test is to provide periodic verification of the leaktight integrity of the primary reactor containment, and systems and components which penetrate containment. The tests assure that leakage through containment and systems and components penetrating containment will not exceed the allowable leakage rate values associated with