March 7, 1995

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Mr. Donald F. Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, MO 63166

SUBJECT: AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-30 - CALLAWAY PLANT, UNIT 1 (TAC NO. M90154)

Dear Mr. Schnell:

The Commission has issued the enclosed Amendment No. 94 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated August 4, 1994, and supplemented on October 31, 1994.

The amendment revises Technical Specification Table 3.3-1 and 4.3-1, and Bases pages B 2-8 and B 3/4 4-1. The changes reflect the reanalysis of the boron dilution transient for shutdown modes to address non-conservatisms in the previous event analysis.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by L. Raynard Wharton

L. Raynard Wharton, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures: 1. Amendment No. 94 to License No. NPF-30
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 1995

Mr. Donald F. Schnell
Senior Vice President - Nuclear
Union Electric Company
Post Office Box 149
St. Louis, MO 63166

SUBJECT: AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-30 - CALLAWAY, UNIT 1 (TAC NO. M90154)

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Union Electric Company (UE, the licensee) dated August 4, 1994, and October 31, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-30 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 94 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. UE shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance. The Technical Specifications are to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raynard Wharton

L. Raynard Wharton, Project Manager
Project Directorate III-3
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: March 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 94

OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages, indicated by an asterisk, are also provided to maintain document completeness.

REMOVE	INSERT
B 2-8	B 2-8
3/4 3-4*	3/4 3-4*
3/4 3-5	3/4 3-5
3/4 3-11*	3/4 3-11*
3/4 3-12	3/4 3-12
3/4 3-12a	3/4 3-12a
B 3/4 4-1	B 3/4 4-1
B 3/4 4-2*	B 3/4 4-2*

REACTOR TRIP SYSTEM INTERLOCKS

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), provides a backup block for Source Range Neutron Flux Multiplication, and allows deenergization of the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the single loop Low Flow trip.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and de-energizes the Source Range high voltage power. On decreasing power; the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7 P-10 Input	4	2	3	1	8
or P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
19. Reactor Trip Breakers	2 2	1 1	2 2	1, 2 3*, 4*, 5*	9,12 10
20. Automatic Trip and Interlock Logic	2 2	1 1	2 2	1, 2 3*, 4*, 5*	31 10

TABLE 3.3-1 (Continued)
TABLE NOTATIONS

* Only if the Reactor Trip System breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

** The boron dilution flux multiplication signals may be blocked during reactor startup in accordance with approved procedures.

The provisions of Specification 3.0.4 are not applicable.

Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) The applicable MODES for these channels noted in Table 3.3-3 are more restrictive and, therefore, applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint; or

b. Above the P-6 (Intermediate Range Neutron Flux interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1,2
e. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 11)	N.A.	1,2,3*,4*,5*
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1,2,3*,4*,5*
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(17), R(18)	N.A.	1,2,3*,4*,5*

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* Only if the Reactor Trip System breakers happen to be closed and the Control Rod Drive System is capable of rod withdrawal.

\# The specified 18 month frequency may be waived for Cycle 1 provided the surveillance is performed prior to restart following the first refueling outage or June 1, 1986, whichever occurs first. The provisions of Specification 4.0.2 are reset from performance of this surveillance.

\#\# Below P-6 (Intermediate Range Neutron Flux interlock) Setpoint.

\#\#\#Below P-10 (Low Setpoint Power Range Neutron Flux interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) For Source Range detectors, integral bias curves are obtained, evaluated, and compared to manufacturer's data. For Intermediate Range and Power Range channels, detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. Determination of the loop specific vessel ΔT value should be made when performing the Incore/Excore quarterly recalibration, under steady state conditions.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip Attachments of the Reactor Trip Breakers.

(8) Deleted

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of 1.7 times the count rate within a 10-minute period.

TABLE NOTATIONS

(10) Setpoint verification is not required.

(11) Following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Flux Multiplication test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within 30 seconds.

(13) Deleted

(14) Deleted

(15) The surveillance MODES specified for these channels in Table 4.3-2 are more restrictive and therefore, applicable.

(16) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY[1] of the Undervoltage and Shunt Trip circuits for the Manual Reactor Trip function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit.

(17) Local manual shunt trip prior to placing breaker in service.

(18) Automatic Undervoltage Trip.

---

1 Complete verification of OPERABILITY of the manual reactor trip switch circuitry shall be performed prior to startup from the first shutdown to Mode 3 occurring after August 7, 1992.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the safety analysis DNBR limits during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing decay heat even in the event of a bank withdrawal accident; however, single failure considerations require that three loops be OPERABLE. A single reactor coolant loop provides sufficient heat removal if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) in MODES 3, 4, and 5 provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the transient mitigation capability of the Boron Dilution Mitigation System (BDMS). With no reactor coolant loop in operation in either MODES 3, 4, or 5, boron dilutions must be terminated and dilution sources isolated. The boron dilution analysis in these MODES takes credit for the mixing volume associated with having at least one reactor coolant loop in operation.

The restrictions on starting a reactor coolant pump in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure and prevent a high pressurizer pressure reactor trip during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

The PORVs are equipped with automatic actuation circuitry and manual control capability. Because no credit for automatic operation is taken in the FSAR analyses for MODE 1, 2 and 3 transients where operation of the PORVs has a beneficial impact on the results of the analysis, the PORVs are considered OPERABLE in either the manual or automatic mode. The automatic mode is the preferred configuration, as this provides pressure relieving capability without reliance on operation action.

Correction letter of 9-7-93

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 94 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated August 4, 1994, Union Electric Company (UEC) requested an amendment to Operating License NPF-30, which would revise the Callaway Plant Technical Specifications (TS). Specifically, the proposed changes would (1) refer to "flux multiplication" rather than "flux doubling" in the Bases for the P-6 reactor trip system interlock, in Note ** of Table 3.3-1, and in Note 12 of Table 4.3-1; (2) revise Note 9 of Table 4.3-1 to reflect a setpoint of 1.7 times the neutron count rate, rather than twice the count rate; and, (3) revise Bases 3/4.4.1 to reflect the boron dilution analysis assumption of one reactor coolant pump (RCP) in operation during Modes 3, 4, and 5, and the requirement for administrative controls to isolate dilution sources in these Modes, if no reactor coolant loop is in operation.

These changes are a result of a reanalysis of the Callaway boron dilution event for shutdown Modes 3, 4, and 5. The reanalysis was performed because of several nonconservative assumptions discovered in the previous boron dilution analysis and delineated in NRC Information Notice 93-32 dated April 21, 1993. One concern involved the applicability of a generic inverse count rate ratio (ICRR) curve, rather than a plant-specific curve. Other concerns involved the time delays and the instrument uncertainty associated with the Boron Dilution Mitigation System (BDMS).

Inadvertent boron dilution during refueling Mode 6 is prevented by administrative controls, which isolate the reactor coolant system (RCS) from the potential sources of unborated water. Therefore, Mode 6 boron dilution events are not analyzed for Callaway.

The October 31, 1994, submittal provided supplemental information which did not affect the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The means of causing an inadvertent boron dilution in the Callaway plant are the opening of the primary water makeup control valve and failure of the blend system, either by controller or mechanical failure. The chemical and volume control system (CVCS) and the demineralized reactor makeup water system (RMWS) are designed to limit the potential rate of dilution to values which, with indication by alarms and instrumentation, would allow sufficient time for automatic or operator response (depending on the Mode of operation) to terminate the dilution. An inadvertent dilution from the RMWS may be terminated by closing the primary water makeup control valve. All expected sources of dilution may be terminated by closing isolation valves BG-LCV-112B and C in the CVCS. The lost shutdown margin may be regained by the opening of isolation valves BN-LCV-112D and E from the refueling water storage tank (RWST), thus allowing the addition of borated water to the RCS.

The BDMS was developed to detect and mitigate an inadvertent boron dilution event occurring in Mode 3, 4, or 5 before a complete loss of shutdown margin (criticality) occurs. The system monitors the output of the source range neutron flux detectors in discrete one-minute intervals and retains the averaged flux data for up to ten of those intervals. The microprocessor compares the average flux value in the most recent interval to each of the prior nine intervals and actuates an alarm and automatic mitigation functions when the flux multiplication setpoint specified in the TS is reached. These automatic actions include aligning the suction of the charging pumps to the RWST and isolating the suction path from the volume control tank (VCT) in order to reborate the RCS.

The Callaway boron dilution analysis of record was performed by Westinghouse in support of Cycle 4 operation and credited the BDMS. However, no uncertainty was assumed for the flux multiplication setpoint of two (flux doubling). In addition, the H. B. Robinson Cycle 3 ICRR data was used for the limiting ICRR versus RCS boron concentration curve for the Callaway analysis. Subsequently, after identifying nonconservatisms in the analysis for another plant relying on a BDMS for mitigating the boron dilution event from shutdown Modes 3, 4, and 5, Westinghouse informed the licensee that the Callaway boron dilution event should be reanalyzed for the current Cycle 7 to address time delays associated with the microprocessor algorithm and signal processing, instrument uncertainty, and concerns regarding the applicability of the assumed ICRR curve.

The boron dilution event reanalysis for Modes 3, 4, and 5 assumed a reduced flux multiplication setpoint of 1.7 (instead of 2.0). Uncertainties have been determined for source range count rates greater than 10 counts per second (cps) and for count rates less than 10 cps. For each estimate of the uncertainty, a safety analysis limit (SAL) was determined presuming a nominal plant setpoint of 1.7 times the reference flux. For count rates greater than 10 cps, an SAL of 2.14 was determined to account for a 25.5% instrument uncertainty. For count rates less than 10 cps, an SAL of 2.65 was used to account for a 55.5% instrument uncertainty.

The dilution flow rate is limited by piping system friction losses and the capacity of two makeup water pumps to supply 260 gpm of unborated water in Modes 3 and 4. During Mode 5, dilution flow is limited by a flow orifice at the RMWS-CVCS system interface to 150 gpm of unborated water. Valve BG-V-0178 in the CVCS is closed during Mode 5 to block unborated water from reaching the RCS.

The initial and critical boron concentration values were revised. These values are based on plant shutdown margins of 1.3% $\Delta k/k$ (Modes 3 and 4) and 1.0% $\Delta k/k$ (Mode 5) and are acceptable since they bound the maximum values used in the Cycle 7 reload evaluation.

A minimum RCS mixing volume of 8995 ft^3 was used for Modes 3, 4, and 5. This corresponds to the active volume of the RCS with one reactor coolant pump in operation and conservatively neglects any mixing in the upper head region. Administrative controls have been put in place by the licensee to ensure that dilution sources are isolated if no reactor coolant loop is in operation. This administrative requirement has also been incorporated into TS Bases 3/4.4.1. The mixing volume assumption is, therefore, acceptable.

The Callaway plant-specific Cycle 5 ICRR versus boron concentration data will be used rather than the H. B. Robinson Cycle 3 ICRR data. This curve was obtained during routine inverse count rate plots by diluting to critical, rather than withdrawing control rods to critical and is, therefore, more appropriate for dilution analysis. The licensee has determined that the Callaway Cycle 5 ICRR data is applicable for the current Cycle 7 and its use is, therefore, acceptable. The licensee, of course, must verify that this ICRR curve is applicable in each cycle-specific reload evaluation.

The following delay times were included in the reanalysis for Cycle 5:

(1) a 10-second signal delay from the microprocessor to mitigation actuation;

(2) a 15-second delay in opening of the CVCS isolation valves from the RWST (BN-LCV-112D and E);

(3) a 10-second delay in closure of the CVCS outlet isolation valves from the VCT (BG-LCV-112B and C), and;

(4) a 131-second delay (Modes 3 and 4) and a 212-second delay (Mode 5) for purge of the CVCS piping from the RWST to the RCS.

The RWST and VCT isolation valves have a 10-second maximum stroke time limit in the Callaway Inservice Testing Program for pumps and valves. The extra 5 seconds in stroke time assumed for the RWST valves represent calculational margin. The CVCS purge delay time is calculated by dividing the purge volume by the appropriate dilution flow rate. The licensee has verified that the dilution flow rates, 150 gpm for Mode 5 and 260 gpm for Modes 3 and 4, are conservative by plotting the calculated RMWS resistances against the reactor makeup water pump performance curve.

Therefore, the NRC staff considers these delay times to be acceptable. They result in a total delay time, from when the BDMS indicates that the setpoint has been reached until reboration of the RCS commences, of 166 seconds for Modes 3 and 4 and 247 seconds for Mode 5.

The results of the new analysis indicate that there is sufficient time for BDMS action to prevent a complete loss of shutdown margin. The minimum departure from nucleate boiling ratio (DNBR) remains well above the safety limit value, power peaking factors and peak local power density remain below allowable limits, and no overpressurization occurs. Therefore, there are no fuel failures and the results are acceptable. The reanalysis of the boron dilution event from shutdown Modes 3, 4, and 5, which incorporate time delays and uncertainties in the BDMS and use the Callaway plant-specific ICRR data, show that acceptable results are attained. Therefore, the proposed TS changes, which are needed to ensure that the analysis assumptions are properly reflected, are acceptable.

Although administrative controls have been established during Mode 6 to isolate the RCS from the potential source of unborated water by locking closed valves, a new source of unborated water has recently been discovered. This source is associated with the use of reactor makeup water (RMW) to rinse items removed from the refueling pool and to spray down the refueling pool walls during the pool drain evolution to facilitate decontamination activities. The refueling pool is connected to both the RCS and the spent fuel pool during refueling operations. In a separate licensing action, the staff is currently reviewing the implementation of additional procedural controls to minimize the amount of unborated water, which could enter the reactor vessel from these washdown procedures.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 49439). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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