November 8, 1993

Docket No. 50-483

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

Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. M87072)

The Commission has issued the enclosed Amendment No. 84 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated June 4, 1993, as clarified by letter dated October 19, 1993.

The amendment revises Technical Specification Tables 2.2-1 and 4.3-1 and associated Bases 3/4.2.2 and 3/4.2.3 by changing the axial flux difference (AFD) penalty function f_1 (delta - I) defined in Note 1 of Table 2.2-1 for the Overtemperature Delta-T reactor trip.

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/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 84 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDIII-3	JBH for PM:PDIII-3	D:PDIII-3	OGC-OWF
MRushbrook	LRWharton:sw	JHannon	
10/25/93	10/25/93 LRW 11/1/93	10/30/93	11/4/93

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

November 8, 1993

Docket No. 50-483

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

Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. M87072)

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Sincerely,

L. Raynard Wharton

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

Enclosures:
1. Amendment No. 84 to
   License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

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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. 84
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 June 4, 1993, and clarified October 19, 1993, 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. 84 , 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/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: November 8, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 84

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 contains marginal lines indicating the area of change. The corresponding overleaf page is also provided to maintain document completeness.

REMOVE	INSERT
2-7	2-7
2-8	2-8
2-9	2-9
3/4 3-12	3/4 3-12
B3/4 2-4	B3/4 2-4
B3/4 2-5	B3/4 2-5

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left\{ K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)} \left[T \left(\frac{1}{1+\tau_6 S}\right) - T' \right] + K_3 (P - P') - f_1(\Delta I) \right\}$$

Where:			
	ΔT	=	Measured ΔT;
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	Lead-lag compensator on measured ΔT;
	τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
	$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
	τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
	ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
	K_1	=	1.15;
	K_2	=	0.0251/°F;
	$\frac{1+\tau_4 S}{1+\tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
	τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \geq 28$ s, $\tau_5 \leq 4$ s;
	T	=	Average temperature, °F;
	$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
	τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.4°F (Referenced T_{avg} at RATED THERMAL POWER);
K_3	=	0.00116/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between -23% and +10%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and botom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that $q_t - q_b$ is more negative than -23%, the ΔT Trip Setpoint shall be automatically reduced by 3.25% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +10%, the ΔT Trip Setpoint shall be automatically reduced by 2.973% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.3% of ΔT span.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)}\left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right)\left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where:

ΔT	=	Measured ΔT;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \geq 8$ s, $\tau_2 \leq 3$ s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, $\tau_3 = 0$ s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.080;
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature;
$\frac{\tau_7 S}{1 + \tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constant utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.0065/°F for T > T" and K_6 = 0 for T $\leq$ T";
T	=	Average Temperature, °F;
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 588.4°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of ΔT span.

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 I 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) 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 twice the count rate within a 10-minute period.

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3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position.

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6.

c. The control rod insertion limits of Specification 3.1.3.6 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured (i.e., inferred), no additional allowances are necessary prior to comparison with the limits of Section 3.2.3. An error allowance of 4% has been included in the limits of Section 3.2.3.

Specifications 3.2.2 and 3.2.3 contain the F_Q and F-delta-H limits applicable to VANTAGE 5 fuel.

Margin between the safety analysis DNBR limits (1.61 and 1.69 for the VANTAGE 5 thimble and typical cells) and the design DNBR limits (1.33 and 1.34 for the VANTAGE 5 thimble and typical cells, respectively) is maintained. A fraction of this margin is utilized to accommodate the

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

appropriate fuel rod bow DNBR penalty (1.3% per WCAP-8691, Rev. 1), the flow anomaly penalty (3.3%), the core vs. pressurizer outlet loss coefficient penalty since the core to pressurizer pressure drop is less than 30 psi as assumed in the VANTAGE 5 transition core safety analyses (0.22%), and a penalty (4%) associated with the Overtemperature Delta-T AFD penalty function deadband assessed for those events in FSAR Chapter 15 that credit the Overtemperature Delta-T reactor trip function. The margin between design and safety analysis DNBR limits of 17.4% for VANTAGE 5 fuel is available for plant design flexibility.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either Normal Operation or RESTRICTED AFD OPERATION, $W(z)_{NO}$ or $W(z)_{RAFDO}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$, is met. $W(z)_{NO}$ accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{RAFDO}$ accounts for the more restrictive operating limits required by RESTRICTED AFD OPERATION which result in less severe transient values. The W(z) functions are specified in the Core Operating Limits Report per Specification 6.9.1.9.

Provisions to account for the possibility of decreases in margin to the $F_Q(z)$ limit during intervals between surveillances are provided. Any decrease in the minimum margin to the $F_Q(z)$ limit compared to the minimum margin determined from the previous flux map is determined by comparing the ratio of:

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

taken from the current map to the same ratio from the previous map. The ratios to be compared from the two flux maps do not need to be calculated at identical z locations. Increases in this ratio indicate that the minimum margin to the $F_Q(z)$ limit has decreased and that additional penalties must be applied to the measured $F_Q(z)$ to account for further decreases in margin that could occur before the next surveillance. More frequent surveillances may also be substituted for the additional penalty.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A

3/4.2.4 QUADRANT POWER TILT RATIO (Continued)

limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain the safety analysis DNBR limit throughout each analyzed transient. The indicated T_{avg} value of 592.6°F and the indicated pressurizer pressure value of 2220 psig correspond to analytical limits of 595.2°F and 2202 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of Section 3.2.5. A measurement uncertainty of 2.2% (including 0.1% for feedwater venturi fouling) for RCS total flow rate has been allowed for in determination of the design DNBR value. The measurement uncertainty for the RCS total flow rate is based upon performing a precision heat balance and using the result to normalize the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, an inspection is performed on the feedwater venturi each refueling outage.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application for license amendment dated June 4, 1993, and clarifying information dated October 19, 1993, Union Electric Company (the licensee), requested changes to Technical Specifications (TS) Tables 2.2-1 and 4.3-1 and associated Bases 3/4.2.2 and 3/4.2.3 for the Callaway Plant, Unit 1. The amendment would change TS Tables 2.2-1 and 4.3-1 and their bases. These changes will revise the axial flux difference (AFD) penalty function, as defined in Table 2.2-1 for the overtemperature delta-T (OTDT) reactor trip functional unit. These changes to the penalty function deadband and the positive power reduction slope will be accommodated using available margin in the departure from nucleate boiling (DNBR). These changes are reflected in the modifications made to the Bases 3/4.2.2 and 3/4.2.3, and by the reductions in the recalibration tolerances (from 3% to 2%) for the incore-vs-excore AFD comparisons surveillance in Note 3 of Table 4.3-1.

In addition, the time constant definitions for OTDT and the over power delta-T (OPDT) reactor trip functions in Tables 2.2-1 will be modified to include inequality signs. These inequality signs indicate the conservative direction for setting these time constants and are the same as those previously approved for Vogtle, Units 1 and 2.

The October 19, 1993, submittal provided clarifying information which did not affect the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Calculation of OTDT AFD Penalty Function

The licensee calculated the OTDT AFD penalty function using the core thermal limits and axial offset limits as presented in the reload safety analysis checklist. These axial offset limits are envelopes of allowable power bands, based on the DNBR limit, and are calculated based on predetermined axial power shapes from various Conditions I and II events of the American Nuclear Society (ANS). Axial offset is a measure of the axial core power distribution in the core, and is defined as the ratio of the percent of rated thermal power generated in the top half of the core to that percent of rated thermal power generated in the bottom half of the core.

Analysis by the licensee indicated that the slope of the positive wing of the penalty function may not provide sufficient penalty for the axial power shapes that have large positive AFD (i.e., top-skewed power shapes). Currently, the penalty function is calculated using a linear extrapolation of the core thermal limits and axial offset limits above 118 percent rated thermal power. The same analysis showed that the slope of the positive wing of the AFD penalty function may not be as high with this approach as it would be if the calculations were performed using actual axial offset data at the higher power levels.

The negative wing of the AFD penalty function, the power reduction slope, is unchanged (shifted slightly from -24% to -23%) because of the large negative AFD deadband and because axial power shapes with large negative axial offsets are not limiting for either Cycle 6 or Cycle 7.

At Callaway, Unit 1, the beginning of cycle life (BOC) axial flux difference under relaxed axial offset control (RAOC) operation can be as high as +12 percent, at 100 percent rated thermal power. However, OTDT trip settings require trip set point reductions if the AFD exceeds +6 percent. In Cycle 6, this condition imposed operational limitations and the licensee expects similar limitations for Cycle 7. To prevent this limiting situation from arising, the licensee determined that the OTDT AFD penalty function deadband, for which there is no trip set point reduction, can be moved out to +10 percent AFD if the negative wing of the AFD penalty function is revised to impose a penalty below -23 percent AFD of 4 percent of the available DNBR margin. This 4 percent will be reduced in Cycle 7 with the reinsertion of thimble plugs (a reduction in the DNBR penalty of 3.1 percent) so that the total DNBR penalty will be less than 1 percent.

2.2 Time Constants and Inequality Signs

The licensee has added inequality signs to the time constant definitions in Notes 1 and 3 of Table 2.2-1 for the OTDT and OPDT reactor trip functions. These additions will add more conservatism to the time constant settings.

The reactor trip system response time, beginning at the time the measured delta-T exceeds the trip function set point at the resistance temperature detectors (RTDs) as defined in the Technical Specifications, will be unaffected. The addition of the time constant inequality signs, the time between the beginning of a transient until the RTDs sense a delta-T higher than the reactor trip set point will be reduced because the effect of the signal conditioning will be to lower the trip set point if the time constants are set in accordance with the conservative directions of the inequality signs. The response time of the OTDT and OPDT reactor trip functions will remain within the assumptions used in the accident analyses. The analyses of the events that credit the OTDT reactor trip will remain as presented in FSAR Chapter 15 and WCAP-10961-P.

The staff has reviewed the detailed analysis submitted by the licensee regarding the inclusion of these inequalities to the appropriate time constants in Table 2.2-1 and finds them acceptable.

The NRC staff has reviewed the reports submitted by the Licensee for the operation of Callaway, Unit 1, cycle 7 and finds that appropriate material was submitted to justify Technical Specification and Bases changes pertaining to the OTDT and OPDT. Based on this review, we have concluded that the requested TS and bases changes satisfy staff positions and requirements in these areas.

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 (58 FR 46240). 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.

Principal Contributors: A. Attard

Date: November 8, 1993