August 21, 1995

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

SUBJECT: AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-30 - CALLAWAY PLANT, UNIT 1 (TAC NO. M92051)

Dear Mr. Schnell:

The Commission has issued the enclosed Amendment No. 102 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 April 17, 1995, as supplemented on June 30, 1995.

The amendment revises Technical Specification 2.2.1, Table 2.2-1. The changes address reducing repeated alarms and partial reactor trips by revising the Overpower Delta-T (OPΔT) setpoint function.

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, 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. 102 to License No. NPF-30
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\CALLAWAY\CAL87073.AMD
* See previous concurrence
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 1995

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

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Sincerely,

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Division of Reactor Projects - III/IV
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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. 102
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 April 17, 1995, as supplemented June 30, 1995, 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.102 , 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: August 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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.

REMOVE	INSERT
2-4	2-4
2-7	2-7
2-9	2-9
2-10	2-10
B 2-5	B 2-5
B 2-6	B 2-6
B 2-6a	B 2-6a

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	SENSOR ERROR Z	(S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N. A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤112.3% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4	0.5	0	≤4% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. Deleted					
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	$\leq 10^5$ cps	$\leq 1.6 \times 10^5$ cps
7. Overtemperature ΔT	9.3	6.47	1.83 +1.24***	See Note 1	See Note 2
8. Overpower ΔT	5.0	1.90	1.65	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0	2.21	2.0	≥1885 psig	≥1874 psig
10. Pressurizer Pressure-High	7.5	4.96	1.0	≤2385 psig	≤2400 psig
11. Pressurizer Water Level-High	8.0	2.18	2.0	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.38	0.6	≥90% of loop minimum measured flow**	≥88.8% of loop minimum measured flow**

* RTP = RATED THERMAL POWER

** Minimum Measured Flow = 95,660 gpm

*** Two Allowances (temperature and pressure, respectively)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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 $\Delta T, \tau_1 \geq 8s, \tau_2 \leq 3s$;
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0s;
ΔT_0	=	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}, $\tau_4 \geq 28s$, $\tau_5 \leq 4s$;
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, τ_6 = 0s;

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 $\Delta T, \tau_1 \geq 8s, \tau_2 \leq 3s$;
$\frac{1}{1+\tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_4	=	1.090;
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}, $\tau_7 \geq 10s$;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE: 3 (Continued)

K_6 = 0.0015/°F for T > T" and K_6 = 0 for T ≤ T";

T = Average Temperature, °F;

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 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 2.4% of ΔT span.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors, and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor Trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Delta-T_o, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. For the startup of a refueled core until measured at 100% Rated Thermal Power (RTP), Delta T_o is initially assumed at a value which is conservatively lower than the last measured 100% RTP Delta T_o for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in vessel ΔT can arise due to several factors, the most prevalent being measured RCS loop flows greater than Minimum Measured Flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific vessel ΔT values. Accurate determination of the loop specific vessel ΔT value should be made when performing the Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by Xe or other transient conditions).

The time constants utilized in the lag compensation of measured ΔT, τ_3, and measured T_{avg}, τ_6, are set in the field at 0 seconds. This setting corresponds to the 7300 NLL cards used for lag compensation of these signals. Safety analyses that credit Overtemperature ΔT for protection must account for these field adjustable lag cards as well as all other

Overtemperature ΔT (Continued)

first order lags (i.e., the combined RTD/thermowell response time and the scoop transport delay and thermal lag). The safety analyses use a total first order lag of less than or equal to 6 seconds.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux Trip.

The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Delta-T_0, as used in the Overtemperature and Overpower ΔT trips, represents the 100% RTP value as measured by the plant for each loop. For the startup of a refueled core until measured at 100% Rated Thermal Power (RTP), Delta T_0 is initially assumed at a value which is conservatively lower than the last measured 100% RTP Delta T_0 for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in vessel ΔT can arise due to several factors, the most prevalent being measured RCS loop flows greater than Minimum Measured Flow, and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific vessel ΔT values. Accurate determination of the loop specific vessel ΔT value should be made when performing the Incore/Excore quarterly recalibration and under steady state conditions (i.e., power distributions not affected by Xe or other transient conditions).

The time constants utilized in the lag compensation of measured ΔT, τ_3, and measured T_{avg}, τ_6, are set in the field at 0 seconds. This setting corresponds to the 7300 NLL cards used for lag compensation of these signals. Safety analyses that credit Overpower ΔT for protection must account for these field adjustable lag cards as well as all other first order lags (i.e., the combined RTD/thermowell response time and the scoop transport delay and thermal lag). The safety analyses use a total first order lag of less than or equal to 6 seconds.

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own Trip Setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated April 17, 1995, Union Electric Company (UE), requested an amendment to Operating License NPF-30, which would change the Technical Specifications (TS) for the Callaway Plant, Unit 1. The proposed amendment would revise TS 2.2.1, Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," and modify the Overpower Delta-T (OPΔT) setpoint. The proposed changes are requested to reduce repeated alarms and partial reactor trips associated with the C-4 control system interlock and OPΔT rod stop that have occurred in past cycles. The C-4 interlock blocks control rod withdrawal and initiates turbine runback when its 2/4 logic coincidence is satisfied. Typically, one channel will exceed the C-4 setpoint for its loop partially satisfying the C-4 logic requirement and resulting in an alarm on that channel. An upper plenum flow anomaly causes a temperature gradation which results in T_{avg} greater than nominal for one loop and the C-4 setpoint being exceeded for that loop. The licensee has proposed changes to the setpoint based on reanalysis.

The June 30, 1995, submittal provided supplemental information which did not affect the initial proposed no significant hazards determination.

2.0 EVALUATION

Callaway has had a flow anomaly in the upper plenum of the reactor vessel since Cycle 2. The anomaly is characterized by a periodic, opposing step changes in loops 2 and 3 hot leg temperatures. During the anomaly transient, hotter water from the center of the core is redirected towards loop 2 while cooler water form the periphery of the core is redirected towards loop 3. Callaway has experienced 1.5°F temperature swings occurring 3-4 times an hour. The frequency and magnitude of the temperature swings vary from cycle to cycle and are not necessarily constant during a particular cycle. At the beginning of Cycle 7 (November 28, 1993 to March 26, 1994) the average alarm rate was 1 alarm/day, however, on occasion as many as 7 alarms were received in one day.

Proposed changes are to the Total Allowance (TA), K_4, K_6, and the Allowable Value in Table 2.2-1. Westinghouse, the licensee's vendor, has confirmed that the revised OPΔT function, in conjunction with the Overtemperature Delta-T (OTΔT) and other reactor protection system functions, will continue to ensure that the core thermal limit lines are adequately protected. In addition, the licensee has stated that their analysis indicates that the maximum core thermal power for all American Nuclear Society Conditions II events will remain below 118.52% of rated thermal power as was previously evaluated for Callaway. While performing the evaluation to justify the changes in the OPΔT setpoint function, Westinghouse determined that it was necessary to reanalyze the full power steam line break coincident with rod withdrawal transient.

This transient was reanalyzed using the LOFTRAN computer code which simulates the neutron kinetics, reactor coolant system (RCS), pressurizer relief and safety valves, pressurizer spray, steam generator and steam generator safety valves. The code computes pertinent plant variables, including temperatures, pressures and power level. Upon initiation of the steamline break, control rods begin to withdraw. The rod withdrawal causes an increase in reactor power and core heat flux to the point at which the OPΔT trip setpoint is reached. Then a reactor trip occurs and control rods insert. The most limiting part of this transient occurs immediately prior to the reactor trip. A departure from nucleate boiling analysis determined that departure from nucleate boiling ratio was greater than the safety analysis limit values at all times during the transient. Additionally, the peak linear heat rate and RCS pressure remained below their respective limits.

The staff has determined that the analyzed results were acceptable; therefore, the proposed changes are acceptable.

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 (60 FR 24922). 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 Contributor: M. Chatterton

Date: August 21, 1995