MAY 04 1987

Docket No. STN 50-483

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

Dear Mr. Schnell:

SUBJECT: CALLAWAY PLANT, UNIT 1 - AMENDMENT NO.22TO FACILITY OPERATING LICENSE NPF-30

The Commission has issued the enclosed Amendment No. 22 to Facility Operating License NPF-30 for the Callaway Plant, Unit 1. The amendment consists of a change to the Technical Specifications (TS) in response to your application dated April 16, 1987.

The amendment revises Table 3.3-5 of the TS to increase the Engineered Safety Features (ESF) response times by fifteen seconds for Items: 2.a. (Containment Pressure-High-1, Safety Injection); 3.a. (Pressurizer Pressure-Low, Safety Injection); and 4.a. (Steam Line Pressure-Low, Safety Injection). These TS revisions are being issued before the expiration of the notice period to preclude any unnecessary delay in plant startup from the current outage. The amendment is effective as of its date of issuance.

A copy of the related Safety Evaluation is enclosed. Notice of issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/

Thomas W. Alexion, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No. 22 to
   License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

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| Date: | 04/30/87 | 04/30/87 | 4/30/87 | 05/4/87 |

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Docket Files
NRC PDR
Local PDR
PDIII-3 r/f
PDIII-3 p/f
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# UNION ELECTRIC COMPANY

## CALLAWAY PLANT, UNIT 1

## DOCKET NO. STN 50-483

## AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 22
License No. NPF-30

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

   A. The application for amendment filed by Union Electric Company (the licensee) dated April 16, 1987, 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:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 22 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Wigginton, Acting Project Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 4, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 22

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 marginal lines indicating the area of change. Corresponding overleaf pages are provided to maintain document completeness.

| REMOVE PAGES | INSERT PAGES |
|---|---|
| 3/4 3-29 | 3/4 3-29 |
| 3/4 3-30 | 3/4 3-30 |
| 3/4 3-32 | 3/4 3-32 |
| - | 3/4 3-32a |
| B 3/4 3-2 | B 3/4 3-2 |
| - | B 3/4 3-2a (repositioned) |

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 1. Manual Initiation | |
| a. Safety Injection (ECCS) | N.A. |
| b. Containment Spray | N.A. |
| c. Phase "A" Isolation | N.A. |
| d. Phase "B" Isolation | N.A. |
| e. Containment Purge Isolation | N.A. |
| f. Steam Line Isolation | N.A. |
| g. Feedwater Isolation | N.A. |
| h. Auxiliary Feedwater | N.A. |
| i. Essential Service Water | N.A. |
| j. Containment Cooling | N.A. |
| k. Control Room Isolation | N.A. |
| l. Reactor Trip | N.A. |
| m. Emergency Diesel Generators | N.A. |
| n. Component Cooling Water | N.A. |
| o. Turbine Trip | N.A. |
| 2. Containment Pressure-High-1 | |
| a. Safety Injection (ECCS) | $\leq$ 29[7]/27[4] |
| 1) Reactor Trip | $\leq$ 2 |
| 2) Feedwater Isolation | $\leq$ 2[5] |
| 3) Phase "A" Isolation | $\leq$ 1.5[5] |
| 4) Auxiliary Feedwater | $\leq$ 60 |
| 5) Essential Service Water | $\leq$ 60[1] |
| 6) Containment Cooling | $\leq$ 60[1] |
| 7) Component Cooling Water | N.A. |
| 8) Emergency Diesel Generators | $\leq$ 14[6] |
| 9) Turbine Trip | N.A. |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 3. Pressurizer Pressure-Low | |
| a. Safety Injection (ECCS) | $\leq 29^{(7)}/27^{(4)}$ |
| 1) Reactor Trip | $\leq 2$ |
| 2) Feedwater Isolation | $\leq 2^{(5)}$ |
| 3) Phase "A"Isolation | $\leq 2^{(5)}$ |
| 4) Auxiliary Feedwater | $\leq 60$ |
| 5) Essential Service Water | $\leq 60^{(1)}$ |
| 6) Containment Cooling | $\leq 60^{(1)}$ |
| 7) Component Cooling Water | N.A. |
| 8) Emergency Diesel Generators | $\leq 14^{(6)}$ |
| 9) Turbine Trip | N.A. |
| 4. Steam Line Pressure-Low | |
| a. Safety Injection (ECCS) | $\leq 39^{(3)}/27^{(4)}$ |
| 1) Reactor Trip | $\leq 2$ |
| 2) Feedwater Isolation | $\leq 2^{(5)}$ |
| 3) Phase "A" Isolation | $\leq 2^{(5)}$ |
| 4) Auxiliary Feedwater | $\leq 60$ |
| 5) Essential Service Water | $\leq 60^{(1)}$ |
| 6) Containment Cooling | $\leq 60^{(1)}$ |
| 7) Component Cooling Water | N.A. |
| 8) Emergency Diesel Generators | $\leq 14^{(6)}$ |
| 9) Turbine Trip | N.A. |
| b. Steam Line Isolation | $\leq 2^{(5)}$ |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 5. Containment Pressure-High-3 | |
| a. Containment Spray | $\leq 32^{(1)}/20^{(2)}$ |
| b. Phase "B" Isolation | $\leq$ 31.5 |
| 6. Containment Pressure-High-2 | |
| Steam Line Isolation | $\leq 2^{(5)}$ |
| 7. Steam Line Pressure-Negative Rate-High | |
| Steam Line Isolation | $\leq 2^{(5)}$ |
| 8. Steam Generator Water Level-High-High | |
| a. Feedwater Isolation | $\leq 2^{(5)}$ |
| b. Turbine Trip | $\leq$ 2.5 |
| 9. Steam Generator Water Level-Low-Low | |
| a. Start Motor-Driven Auxiliary Feedwater Pumps | $\leq$ 60 |
| b. Start Turbine-Driven Auxiliary Feedwater Pump | $\leq$ 60 |
| 10. Loss-of-Offsite Power | |
| Start Turbine-Driven Auxiliary Feedwater Pump | N.A. |
| 11. Trip of All Main Feedwater Pumps | |
| Start Motor-Driven Auxiliary Feedwater Pumps | N.A. |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 12. Auxiliary Feedwater Pump Suction Pressure-Low | |
| Transfer to Essential Service Water | N.A. |
| 13. RWST Level-Low-Low Coincident with Safety Injection | |
| Automatic Switchover to Containment Sump | $\leq$ 60 |
| 14. Loss of Power | |
| a. 4 kV Bus Undervoltage-Loss of Voltage | $\leq$ 14 |
| b. 4 kV Bus Undervoltage-Grid Degraded Voltage | $\leq$ 144 |
| 15. Phase "A" Isolation | |
| a. Control Room Isolation | N.A. |
| b. Containment Purge Isolation | $\leq 2^{(5)}$ |

TABLE NOTATIONS

(1) Diesel generator starting and sequence loading delays included.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Does not include valve closure time.

(6) Includes time for diesel to reach full speed.

TABLE NOTATIONS (Continued)

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

## 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service times for the Reactor Protection Instrumentation System," supplements to that report, and the NRC's Safety Evaluation dated February 21, 1985. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor Trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

Engineered Safety Features response time specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 3 and 4) are based on values assumed in the non-LOCA safety analyses. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 7), the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response time specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trips, (3) Feedwater System isolates, (4) the emergency diesel generators start, (5) containment spray pumps start and automatic valves position, (6) containment isolates, (7) steam lines isolate, (8) Turbine trips, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, (11) essential service water pumps start and automatic valves position, and (12) isolate normal control room ventilation and start Emergency Ventilation System.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## RELATED TO AMENDMENT NO. 22 TO FACILITY OPERATING LICENSE NO. NPF-30

## UNION ELECTRIC COMPANY

## CALLAWAY PLANT, UNIT 1

## DOCKET NO. STN 50-483

### INTRODUCTION

By letter dated April 16, 1987, Union Electric Company (the licensee) informed the staff that Westinghouse Electric Corporation recently discovered that it had assumed simultaneous, rather than sequential, operation of valves in the calculation of time it takes to get a safety injection (SI) of 2000 ppm borated water into the Reactor Coolant System (RCS). Since the valves that transfer the charging pump suction from the Volume Control Tank (VCT) to the Refueling Water Storage Tank (RWST), which contains 2000 ppm boron, are operated sequentially, it was found that safety injection (ECCS) response times listed in Table 3.3-5 of the Technical Specifications (TS) were not achievable. There were too short by the 15-second delay encountered by the sequential operation of the two valves.

### EVALUATION

The primary function of the ECCS is to supply water to the RCS in the event of a loss of coolant accident (LOCA). Since a LOCA is not a reactivity induced accident, the 2000 ppm boron is not immediately needed. It is only needed to maintain subscriticality in the long term. Therefore, for those SI actuation signals that are only intended to provide protection against a LOCA, this 15-second delay in the delivery of 2000 ppm borated water has no effect on the safety analysis.

The only non-LOCA transient impacted by this increased response time is the steam line break event. No other FSAR Chapter 15 transient takes credit for short-term boration from the RWST.

The licensee compared calculations of the steam line break accident with and without the additional SI delay. The calculations showed no significant change in the consequences. One of the reasons for this is that the additional delay occurs early in the steam line break event when the RCS pressure is high and the SI flow rate is relatively small. In addition, the licensee stated that studies of the steam line break accident have generally shown that the consequences are not sensitive to large changes in SI flow or boron concentration. The licensee, therefore, concluded that the departure from nucleate boiling design basis for the steam line break analysis is still met and that the conclusions presented in the FSAR remain valid.

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The staff has reviewed the licensee's analysis and finds the licensee's conclusions acceptable. Thus, the staff concludes that the requested technical specification changes satisfy the applicable regulatory requirements and are acceptable.

EMERGENCY CIRCUMSTANCES

These TS changes are being issued before the expiration of the notice period to prelcude an unnecessary delay in plant startup from the current outage. On April 13, 1987, the licensee received a letter from Westinghouse informing them of a potential issue concerning the time required to change charging pump suction from the VCT to the RWST following a postulated steam line break event (see introduction section). The licensee completed their own evaluation and determined on April 15, 1987 that the issue was indeed applicable to Callaway Plant and that changes to the TS were needed for plant startup from the current outage. The licensee then promptly notified the staff (on April 15, 1987) of the situation at Callaway and followed-up with a license amendment application dated April 16, 1987.

The Commission has determined that emergency circumstances exist in that swift action is necessary to avoid a delay in startup not related to safety and finds that for the reason stated above, and an accelerated outage schedule, emergency circumstances exist.

In connection with a request indicating an emergency, the Commission expects its licensees to apply for license amendments in a timely fashion. However, with this consideration in mind, it has been determined that a circumstance has arisen where the licensee and the Commission must act quickly, and the licensee has made a good effort to make a timely application.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

In accordance with 10 CFR 50.92, the Commission may make a final determination that a license amendment involves no significant hazards consideration if operation of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The information in this section provides the staff's evaluation of this license amendment against the three criteria.

The staff has confirmed the basis of the no significant hazards findings described in the notice published in the Federal Register on April 22, 1987 (52 FR 13367). The amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated. An increase in the acceptance criterion for the ESF response time is acceptable since the evaluation of the impact of the increased delay on the steam line break event demonstrated that the departure from nucleate boiling design basis is still met. The conclusions in the FSAR remain valid.

The amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated. There are no new failure modes associated with this proposed change, as no design changes have been made. No new accident is created because the same equipment is assumed to perform in the same manner as before. Therefore, an increase in the ESF response times for high containment pressure, low pressurizer pressure, and low steam line pressure does not create the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report.

The amendment does not involve a significant reduction in a margin of safety. There is no impact on the consequences on protective boundaries, and all acceptance criteria in the analysis of record are still met. Therefore, the safety limits will still be met.

Therefore, the staff concluded that:

(1) Operation of the facility in accordance with the amendment would not significantly increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the amendment would not involve a significant reduction in a margin of safety.

Therefore, we conclude that the amendment to Facility Operating License No. NPF-30 to support operation of the Callaway Plant, Unit 1, which revises Table 3.3-5 of the TS to increase the response time by 15 seconds for certain SI functions involves no significant hazards considerations.

STATE CONSULTATION

In accordance with the Commission's regulations, consultation was held with the State of Missouri by telephone. The State expressed no concern, either from the standpoint of safety or of our no significant hazards consideration determination.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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. 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. 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.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and 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: T. Alexion, PD33, DRSP,
E. Lantz, SRXB, DEST

Dated: May 4, 1987