

August 5, 1993

Docket No. 50-483

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

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Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. NPF-30
(TAC NO. M82627)

The Commission has issued the enclosed Amendment No. 83 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 December 4, 1991, as clarified by letters dated March 11, 1992, and March 11, 1993.

The amendment revises Technical Specifications 3/4.4.4 and its associated bases and 3.4.9.3 to address changes as recommended in Generic Letter 90-06, "RESOLUTION OF GENERIC ISSUE 70, 'POWER-OPERATED RELIEF VALVE AND BLOCK VALVE RELIABILITY,' AND GENERIC ISSUE 94, 'ADDITIONAL LOW-TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT-WATER REACTORS,' PURSUANT TO 10 CFR 50.54(f)" dated June 25, 1990.

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/V
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 83 to License No. NPF-30
2. Safety Evaluation

cc w/enclosures:
See next page

*See Previous Concurrences			
LA: PDIII-3:DRPW	*PM: PDIII-3:DRPW	*D: PDIII-3:DRPW	*OGC-OWF
PMRushbrook	LRWharton/sw	JHannon	M.Zobler
8/5/93	7/27/93	6/26/93	7/27/93
DOCUMENT NAME:	g:\callaway\CAL82627.AMD		

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

August 5, 1993

Docket No. 50-483

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Senior Vice President - Nuclear
Union Electric Company
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Sincerely,

A handwritten signature in cursive script that reads "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:

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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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for the Environment
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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. 83
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 December 4, 1991, and clarified March 11, 1992, and March 11, 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:

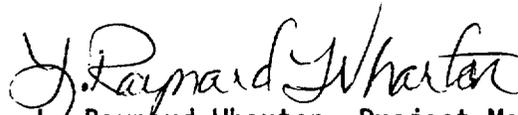
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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 83, 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 date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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: August 5, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 83

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

3/4 4-10

3/4 4-34

3/4 4/35

B 3/4 4-2

B 3/4 4-15

INSERT

3/4 4-10

3/4 4/34

3/4 4/35

B 3/4 4-2

B 3/4 4-15

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with at least two groups of backup pressurizer heaters each having a capacity of at least 150 kW and a water level of less than or equal to 92% (1657 cubic feet).

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With one group of backup pressurizer heaters inoperable, restore at least two groups of backup heaters to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water level shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

- a. With one or both PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status, or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.
- d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both valves are inoperable; restore any remaining inoperable block valve to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.
- e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

*With all RCS cold leg temperatures above 368°F.

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 200°F in any 1-hour period, and
- c. A maximum spray water temperature differential of 583°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following groups of two Overpressure Protection devices shall be OPERABLE when the Reactor Coolant System (RCS) is not depressurized with an RCS vent of the greater than or equal to 2 square inches:

- a. Two residual heat removal (RHR) suction relief valves each with a Setpoint of 450 psig \pm 3%, or
- b. Two power-operated relief valves (PORVs) with Setpoints which do not exceed the limit established in Figure 3.4-4, or
- c. One RHR suction relief valve and one PORV with Setpoints as described above.

APPLICABILITY: MODE 3 when the temperature of any RCS cold leg is less than or equal to 368°F, MODES 4 and 5, and MODE 6 with the vessel head on.

ACTION:

- a. With one of the two required overpressure protection devices inoperable in MODES 3 or 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS with an RCS vent of greater than or equal to 2 square inches within the next 8 hours.
- b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or depressurize and vent the RCS with an RCS vent of greater than or equal to 2 square inches within the next 8 hours.
- c. With both of the two required overpressure protection devices inoperable, depressurize and vent the RCS with an RCS vent of greater than or equal to 2 square inches within 8 hours.
- d. In the event either the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- e. The provision of Specification 3.0.4 are not applicable.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

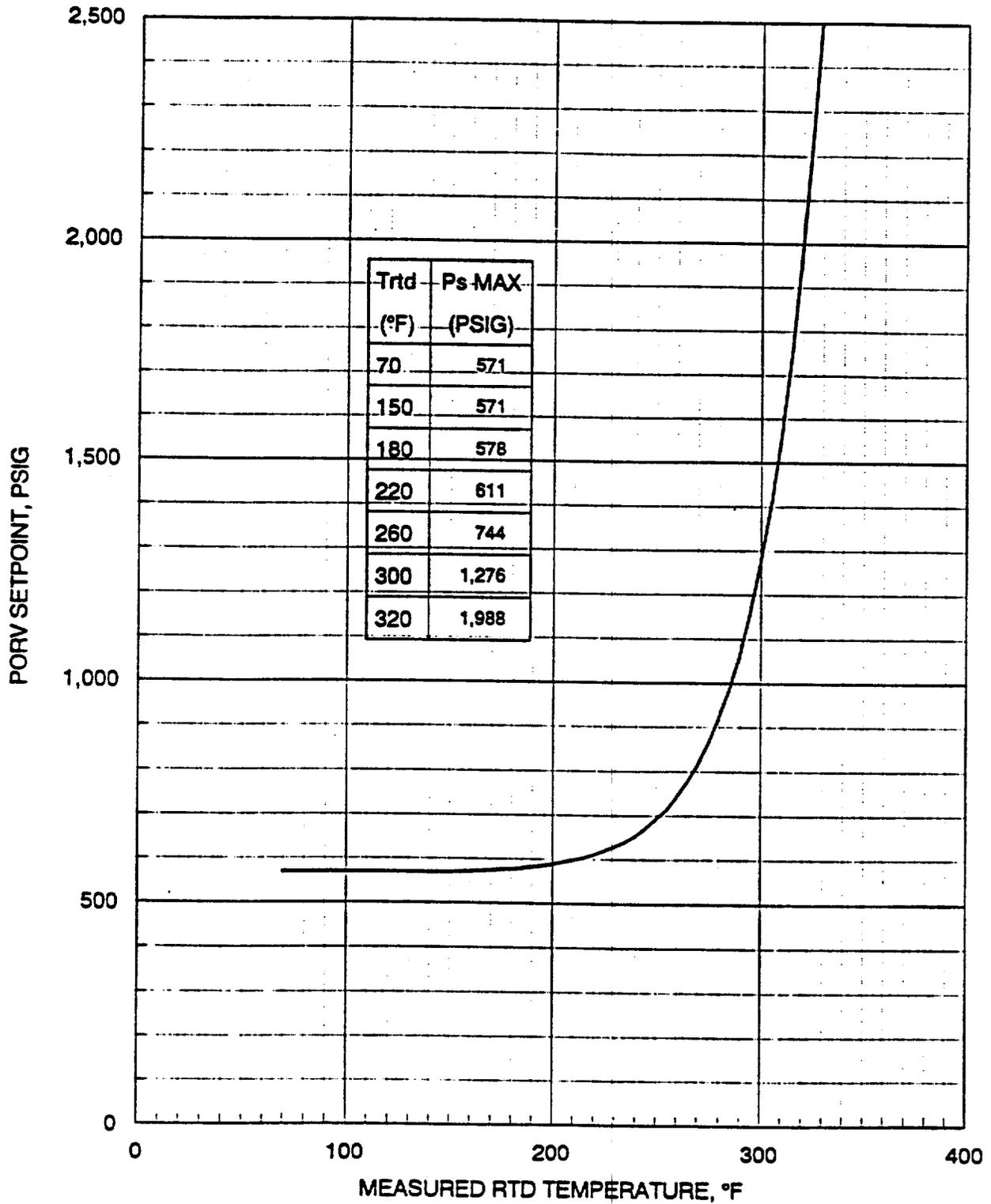
- a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;
- b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and
- c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

- a. For RHR suction relief valve 8708B:
By verifying at least once per 72 hours that RHR RCS suction isolation valves (RRSIV) EJ-HV-8701B and BB-PV-8702B are open.
- b. For RHR suction relief valve 8708A:
By verifying at least once per 72 hours that RRSIV EJ-HV-8701A and BB-PV-8702A are open.
- c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 With the RCS vented, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

**FIGURE 3.4-4
 MAXIMUM ALLOWED PORV SETPOINT
 FOR THE COLD OVERPRESSURE MITIGATION SYSTEM**



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) or one RHR pump 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 capability of operator recognition and control.

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.

REACTOR COOLANT SYSTEM

BASES

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 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.

REACTOR COOLANT SYSTEM

BASES

HEATUP (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two RHR suction relief valves, one RHR suction relief valve and one PORV, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

In addition to opening RCS vents to meet the requirement of Specification 3.4.9.3c., it is acceptable to remove a pressurizer Code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made

REACTOR COOLANT SYSTEM

BASES

COLD OVERPRESSURE (Continued)

possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; 3) instrument uncertainties; and 4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, technical specifications require lockout of both safety injection pumps and all but one centrifugal charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature. Exceptions to these mode requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE and no safety injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis LOCA's occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and safety injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic safety injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of safety injection (one centrifugal charging pump, and one safety injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by operation of both PORV's without exceeding Appendix G limit. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of safety injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both safety injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent safety injection occur above 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F, two RCP and all pressurizer safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the pressurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

On June 25, 1990, the staff issued Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)." The generic letter represented the technical resolution of the above mentioned generic issues.

By letters dated December 4, 1991, as clarified by March 11, 1992, and March 11, 1993, the Union Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The proposed amendment would change the plant Technical Specifications (TSs) based on the recommendations provided by the NRC staff in Generic Letter (GL) 90-06.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection channel in operating modes 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letters dated March 11, 1992, and March 11, 1993, the licensee provided clarifying information regarding full stroke testing of the PORVs, consistent with GL 90-06, that did not change the initial proposed determination of no significant hazards consideration or affect the notice published April 1, 1992 (57 FR 11116).

2.0 EVALUATION

GENERIC ISSUE 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

The proposed changes revise Technical Specification 3/4.4.4 and its associated Bases by clarifying and modifying the limiting conditions for operation of power-operated relief valves. The revisions adequately meet the intent of staff positions provided in the generic letter, with acceptable plant specific alternatives.

The proposed changes to the Callaway technical specifications included in the licensee's letters of December 4, 1991, March 11, 1992, and March 11, 1993, are consistent with that proposed in the staff's generic letter. The proposed changes involve plant operations with the block valves closed due to leaking PORVs and appropriate plant mode for PORV stroke testing. The licensee has adopted the staff position by stipulating power be maintained to closed block valves so it remains operable and may be subsequently opened to allow use for control of reactor coolant system pressure.

The staff has reviewed the licensee's proposed modifications to the Callaway technical specifications. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and justified in the NUREG-1316 regulatory analysis, mentioned above, the staff finds the proposed modifications to be acceptable.

GENERIC ISSUE 94

The actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

The proposed changes revise Technical Specification 3.4.9.3 to reflect the use of either the PORVs or the residual heat removal suction relief valves as overpressure protection devices.

The proposed changes to the Callaway Technical Specifications included in the licensee's letters of December 4, 1991, March 11, 1991, and March 11, 1992, provide a modified alternative to that proposed in the staff's generic letter. However, the modification has been reviewed and found to be consistent with the intent of the staff position on LTOP protection.

The staff has reviewed the licensee's proposed modifications to the Callaway technical specifications. Since the proposed modifications are consistent with the staff's position previously stated in the generic letter and justified in the above mentioned regulatory analysis, the staff finds the proposed modifications to be 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

This amendment changes a requirement with respect to installation or use of a facility component 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 effluent 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 (57 FR 11116). 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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