March 27, 1989

Docket No. 50-483

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

Dear Mr. Schnell:

SUBJECT: AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. 71782)

The Commission has issued the enclosed Amendment No. 42 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 January 6, 1989 as supplemented by a letter dated February 10, 1989.

The amendment reduces the required Residual Heat Removal (RHR) system flowrate during Mode 6 operation, deletes the RHR autoclosure interlock (ACI) function, and allows the safety injection (SI) pumps to be energized with the head on and with water level not above the top of the reactor vessel flange, in Modes 5 and 6.

Copies of the Safety Evaluation and of the notice of issuance are also enclosed. The notice of issuance has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

Thomas W. Alexion, Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
1. Amendment No. 42 to License No. NPF-30
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

Office:	LA/PDIII-3	PM/PDIII-3	OGC	PD/PDIII-3
Surname:	PKreutzer	TAlexion/tg		JHannon
Date:	3/13/89	3/13/89	3/15/89	3/13/89

Mr. D. F. Schnell
Union Electric Company

Callaway Plant
Unit No. 1

cc:

Dr. J. O. Cermack
CFA Inc.
4 Professional Dr., Suite 110
Gaithersburg, MD 20879

Gerald Charnoff, Esq.
Thomas A. Baxter, Esq.
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N. W.
Washington, D. C. 20037

Mr. T. P. Sharkey
Supervising Engineer,
Site Licensing
Union Electric Company
Post Office Box 620
Fulton, Missouri 65251

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1
Steedman, Missouri 65077

Mr. Alan C. Passwater, Manager
Licensing and Fuels
Union Electric Company
Post Office Box 149
St. Louis, Missouri 63166

Manager - Electric Department
Missouri Public Service Commission
301 W. High
Post Office Box 360
Jefferson City, Missouri 65102

Regional Administrator
U. S. NRC, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Ronald A. Kucera, Deputy Director
Department of Natural Resources
P. O. Box 176
Jefferson City, Missouri 65102

Mr. Bart D. Withers
President and Chief
Executive Officer
Wolf Creek Nuclear Operating
Corporation
P. O. Box 411
Burlington, Kansas 66839

Mr. Dan I. Bolef, President
Kay Drey, Representative
Board of Directors Coalition
for the Environment
St. Louis Region
6267 Delmar Boulevard
University City, Missouri 63130

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. STN 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
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 January 9, 1989 as supplemented by a letter dated February 10, 1989, 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. 42, 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

James R. Hall for

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 27, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 42

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	INSERT
3/4 4-35	3/4 4-35
3/4 5-4	3/4 5-4
3/4 5-9	3/4 5-9
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10
B 3/4 4-15	B 3/4 4-15
B 3/4 5-2	B 3/4 5-2
-	B 3/4 5-3
B 3/4 9-2	B 3/4 9-2

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.

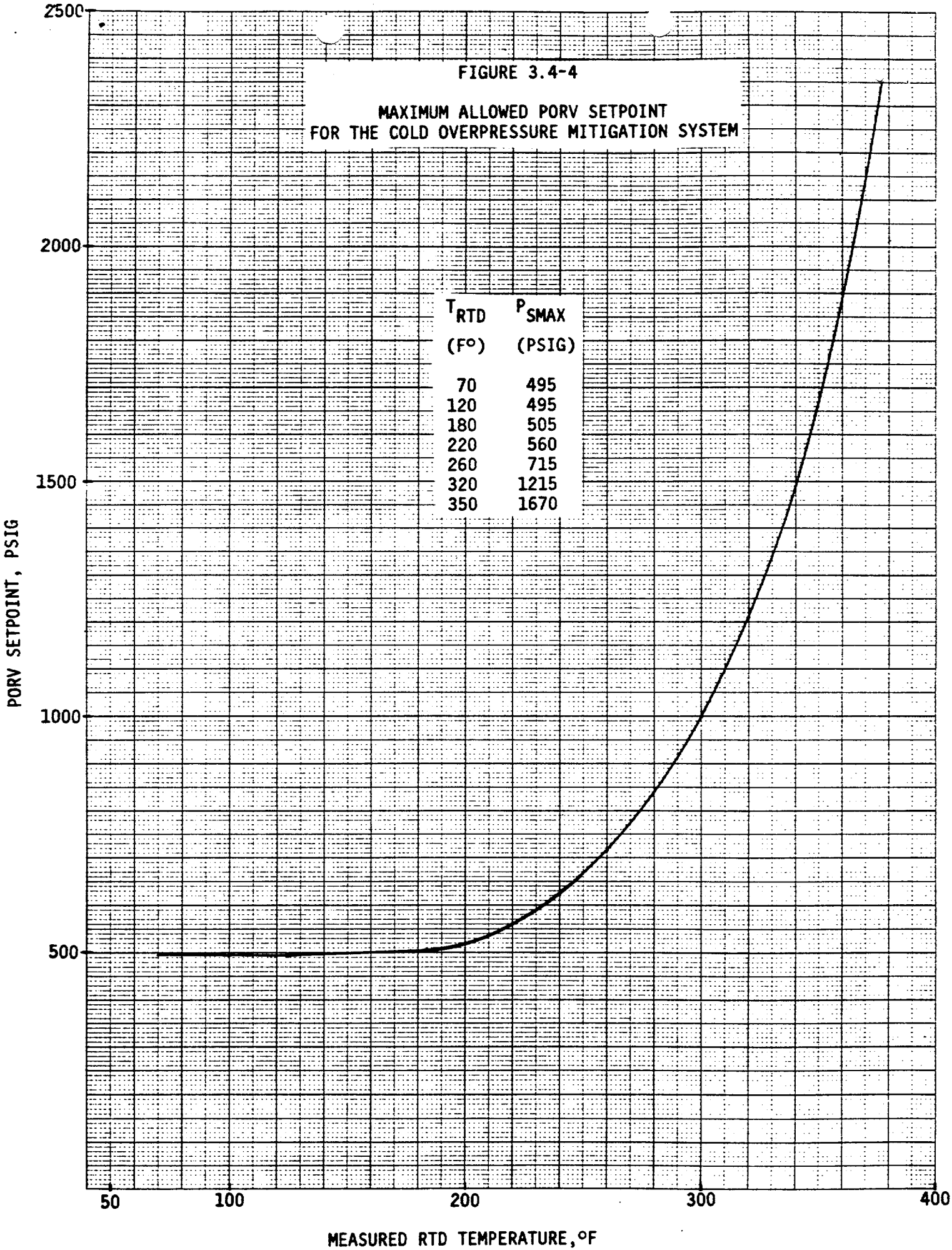
c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT
FOR THE COLD OVERPRESSURE MITIGATION SYSTEM



T_{RTD} (F°)	P_{SMAX} (PSIG)
70	495
120	495
180	505
220	560
260	715
320	1215
350	1670

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and automatically transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump and the Safety Injection pumps declared inoperable pursuant to Specification 4.5.3.2 provided the centrifugal charging pump and the Safety Injection pumps are restored to OPERABLE status within 4 hours prior to the temperature of one or more of the RCS cold legs exceeding 375°F.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
BN-HV-8813	Safety Injection to RWST Isolation Vlv	Open
EM-HV-8802A(B)	SI Pump Discharge Hot Leg Iso Vlvs	Closed
EM-HV-8835	Safety Injection Cold Leg Iso Valve	Open
EJ-HV-8840	RHR/SI Hot Leg Recirc Iso Valve	Closed
EJ-HV-8809A	RHR to Accum Inj Loops 1 & 2 Iso Vlv	Open
EJ-HV-8809B	RHR to Accum Inj Loops 3 & 4 Iso Vlv	Open

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation action of the RHR System from the Reactor Coolant System by ensuring that, with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig, the interlocks prevent the valves from being opened.

3/4.5.4 ECCS SUBSYSTEMS - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.5.4 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 with the water level above the top of the reactor vessel flange, and MODE 6 with the reactor vessel head on and with the water level above the top of the reactor vessel flange.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 31 days.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per twelve hours one RHR loop shall be verified in operation and circulating coolant at a flow-rate of:

a) greater than or equal to 1000 gpm, and

b) sufficient to maintain the RCS temperature at less than or equal to 140°F.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6 when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least once per 12 hours one RHR loop shall be verified in operation and circulating coolant at a flow-rate of:

a) greater than or equal to 1000 gpm, and

b) sufficient to maintain the RCS temperature at less than or equal to 140°F.

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, or two RHR suction relief valves, 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

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 centifugal 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 and in accordance with the schedule in Table 4.4-5.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a MODE where this capability is not required. In order to perform check valve surveillance testing per 4.0.5 or 4.4.6.2.2 above 1000 psig RCS pressure, one accumulator isolation valve may be closed for up to 2 hours in mode 3 only.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve. In addition, the requirement to verify all Safety Injection pumps to be inoperable in MODE 4, in MODE 5 with the water level above the top of the reactor vessel flange, and in MODE 6 with the reactor vessel head on and with the water level above the top of the reactor vessel flange, provides assurance that the mass addition can be relieved by a single PORV or RHR suction relief valve.

With the water level not above the top of the reactor vessel flange and with the vessel head on, Safety Injection pumps may be available to mitigate the effects of a loss of decay heat removal during partially drained conditions.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure, that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensure that a failure of one valve will not cause an intersystem LOCA. The Surveillance Requirement to vent the ECCS pump casings and accessible, i.e., can be reached without personnel hazard or high radiation dose, discharge piping ensures against inoperable pumps caused by gas binding or water hammer in ECCS piping.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

REFUELING WATER STORAGE TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 is sufficient to prevent reactor criticality during refueling operations. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

The OPERABILITY of this system ensures the containment purge penetrations will be automatically isolated upon detection of high radiation levels within containment. The OPERABILITY of this system is required to restrict the release of radioactive materials from the containment atmosphere to the environment.

The restriction on the setpoint for GT-RE-22 and GT-RE-33 is based on a fuel handling accident inside the Containment Building with resulting damage to one fuel rod and subsequent release of 0.1% of the noble gas gap activity, except for 0.3% of the Kr-85 gap activity. The setpoint concentration of 5E-3 uCi/cc is equivalent to approximately 150 mR/hr submersion dose rate.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) manipulator cranes will be used for movement of drive rods and fuel assemblies, (2) each crane has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to maintain a 1000 gpm flowrate ensures that there is adequate flow to prevent boron stratification. The RHR flow to the RCS will provide adequate cooling to prevent exceeding 140°F and to allow flowrates which provide additional margin against vortexing at the RHR pump suction while in partial drain operation.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY
CALLAWAY PLANT, UNIT 1
DOCKET NO. STN 50-483

1.0 INTRODUCTION

By letters dated January 6, 1989 and February 10, 1989 Union Electric Company proposed changes to Technical Specifications (TS's) 4.9.8.1, 4.9.8.2, and the associated Bases to reduce the required Residual Heat Removal (RHR) system flow rate during Mode 6 operation. At the currently required flowrate of 2800 gpm the RHR system could be susceptible to vortexing at the RHR pumps suction piping during Reactor Coolant System (RCS) partial drain operations. Vortexing can lead to RHR system air entrainment and pump cavitation and subsequent loss of RHR system flow.

In addition the licensee proposes to change TS 4.4.9.3.2, 4.5.2.d, and the associated Bases to delete the Autoclosure Interlock function (ACI), and to change TS 3.5.4, and its associated Bases to allow safety injection pumps to be energized with the head on and with the water level not above the top of the reactor vessel flange, in Modes 5 and 6. The pumps may then be used in case of a loss of RHR when the RCS is partially drained.

The most significant change proposed by the licensee is the removal of Autoclosure Interlock. The staff review of this issue has focused on assuring that the changes proposed for Callaway meet the staff position on the removal of the autoclosure interlock as set forth in the staff's safety evaluation for Diablo Canyon, transmitted by letter from Harry Rood (NRC) to J. D. Shiffer (Pacific Gas & Electric), dated February 17, 1988.

2.0 EVALUATION

2.1 Removal of ACI

The staff position taken on removal of ACI at Diablo Canyon consisted of hardware changes and procedural enhancements which the staff believes will produce a net safety benefit compared to the current plant arrangement. The hardware changes consist of the addition of an alarm to each RHR suction valve. The alarm actuates if the valve is open and the pressure is greater than the open permissive setpoint and less than the RHR design pressure minus the RHR pump head pressure. The open permissive which prevents these valves from being opened must be left in place and not be disabled by the

addition of the alarm and the removal of the ACI circuitry. The valve position indicator to the alarm must not be affected by power lockout of the valves and a method independent of the alarm for determining valve position should be available in the control room following power lockout of the RHR suction valves. The procedural modifications required are as follows:

1. The alarm response procedure used during plant startup should be modified to reflect alarm recognition responses for the added alarm. The procedure should be revised to direct the operator to take the necessary actions to close the open RHR suction valve(s), if they are not closed following alarm actuation. If this is not possible, the operator should be instructed to not pressurize further and to return to the safe shutdown mode of operation.

2. A surveillance procedure for the RHR suction valve alarms is added to ensure these alarms remain operable.

3. A method independent of the alarm should be used to ensure that these valves are closed when the power to those valves is locked out. For example, the valves could be leak-checked after power lockout.

Besides the hardware and procedural changes described above, Diablo Canyon and Callaway were requested to review the sizing of the valve operators on the RHR suction valves to ensure that it would be unlikely that these valves could be opened against full system pressure. This provides still another level of protection to ensure the integrity of the high/low pressure system interface.

The staff has reviewed the Callaway submittals and has found that the proposed changes meet the hardware and procedural modifications described above which have been previously approved by the staff for Diablo Canyon and are, therefore, acceptable.

2.2 Reduction In Required RHR System Flowrate

Operation with the RCS partially drained in Modes 5 and 6 is necessary for required inspection and maintenance of RCS components such as reactor coolant pumps and steam generators. As indicated in NUREG-1269, "Loss of Residual Heat Removal at Diablo Canyon Unit 2," reduced RHR flowrate would provide a greater margin against vortexing and preclude an inadvertent loss of decay heat removal capability due to air entrainment and cavitation of the RHR pumps. As the time after plant shutdown increases, decay heat removal requirements from the RHR suction flow are reduced since decay heat decreases as a function of time after initial reactor shutdown. The change proposed by the licensee will provide sufficient flowrate to maintain RCS less than or equal to 140°F. In addition, a minimum RHR flowrate is

required to prevent boron stratification to minimize the potential for localized variation in boron concentration in the RCS. For Callaway, Westinghouse has recommended a minimum flowrate of 1000 gpm. Since the proposed Technical Specification will require that the RHR flowrate is maintained at least greater than 1000 gpm and sufficient to maintain RCS temperature less than 140°F, the staff finds the proposed Technical Specification modifications to be acceptable.

2.3 Allowing Safety Injection Pumps to be Energized in Modes 5 and 6

It is Callaway's intention that during reactor coolant system reduced inventory conditions, the safety injection pump motor circuit breakers will be racked in and secured in the open position by placing the control room handswitch in the pull-to-lock position. This will prevent the safety injection pumps from being inadvertently started by a signal, but will allow the operators to start the pumps from the control room if needed to mitigate a loss of decay heat removal.

This modification was proposed by the staff in Generic Letter 88-17 and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35 an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 20, 1989 (54 FR 11463). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 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; 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.

5.0 ACKNOWLEDGEMENT

Principal Contributor: G. Schwenk, SRXB

Dated: March 27, 1989

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

UNION ELECTRIC COMPANY

DOCKET NO. 50-483

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 42 to Facility Operating License No. NPF-30, issued to Union Electric Company, which revised the Technical Specifications for operation of the Callaway Plant, Unit 1, located in Callaway County, Missouri. The amendment was effective as of the date of issuance.

The amendment modified the Technical Specifications to reduce the required Residual Heat Removal (RHR) system flowrate during Mode 6 operation, delete the RHR autoclosure interlock (ACI) function, and allow the safety injection (SI) pumps to be energized with the head on and with water level not above the top of the reactor vessel flange, in Modes 5 and 6 .

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action was published in the FEDERAL REGISTER on February 8, 1989 (54 FR 6222). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated January 6, 1989, and supplemented by letter dated February 10, 1989, (2) Amendment No. 42 to License No. NPF-30, (3) the Commission's related Safety Evaluation dated March 27, 1989 and (4) the Environmental Assessment dated March 13, 1989. All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street NW, and at the Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251, and the John M. Olin Library, Washington University, Skinker and Lindell Boulevards, St. Louis, Missouri 63130. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects III, IV, V and Special Projects.

Dated at Rockville, Maryland this 27th day of March 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall

James R. Hall, Acting Director
Project Directorate III-3
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation