April 2, 1998

Mr. Garry L. Randolph
Vice President and Chief Nuclear Officer
Union Electric Company
Post Office Box 620
Fulton, Missouri 65251

SUBJECT: CALLAWAY PLANT - AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. NPF-30 (TAC NO. M99918)

Dear Mr. Randolph:

The Commission has issued the enclosed Amendment No. 124 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 17, 1997, as supplemented by letters dated March 3, 1998, and March 17, 1998.

The amendment modifies the plant heatup and cooldown curves and the maximum allowable power operated relief valve setpoint for cold overpressure protection. We approved your request for an exemption from the requirements of 10 CFR 50.60 "Acceptance Criteria for Fracture Prevention for Light Water Nuclear Power Reactors for Normal Operation" in order to apply the American Society of Mechanical Engineers (ASME) Code Case N-514, "Low Temperature Overpressure Protection." The Code case was used in developing the cold overpressure mitigation system setpoints.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original Signed By
Barry C. Westreich, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures: 1. Amendment No. 124 to NPF-42
2. Safety Evaluation

cc w/encls: See next page

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

UNION ELECTRIC COMPANY

CALLAWAY PLANT UNIT 1

DOCKET NO. 50-483

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
License No. NPF-30

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

   A. The application for amendment to the Callaway Plant Unit 1 (the facility) Facility Operating License No. NPF-30 filed by the Union Electric Company (the Company), dated October 17, 1997, as supplemented by letters dated March 3, 1998, and March 17, 1998, 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, as amended, 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 license 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. 124 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance to be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Barry C. Westreich, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 2, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-36	3/4 4-36
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-15	B 3/4 4-15
B 3/4 4-16	B 3/4 4-16
B 3/4 5-2	B 3/4 5-2

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2708-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 100.4°F
3/4T, 84.2 °F

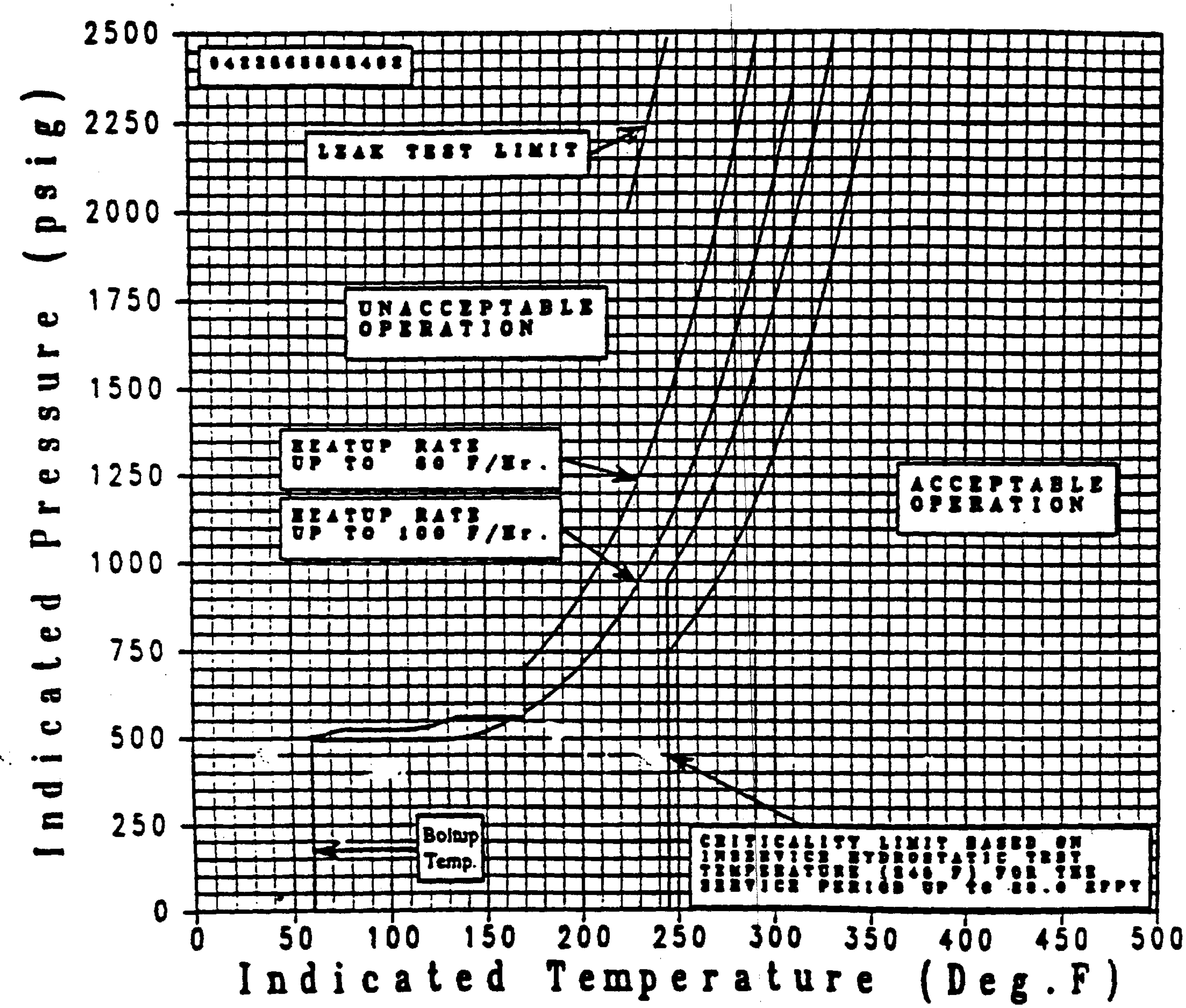


FIGURE 3.4-2

Callaway Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rtes of 60 and 100°F/hr) Applicable for the First 20 EFPY (With Margins for Instrumentation Errors) Includes Vessel flange requirements of 170°F and 561 psig per 10 CFR 50, Appendix G.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2708-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 100.4°F
3/4T, 84.2 °F

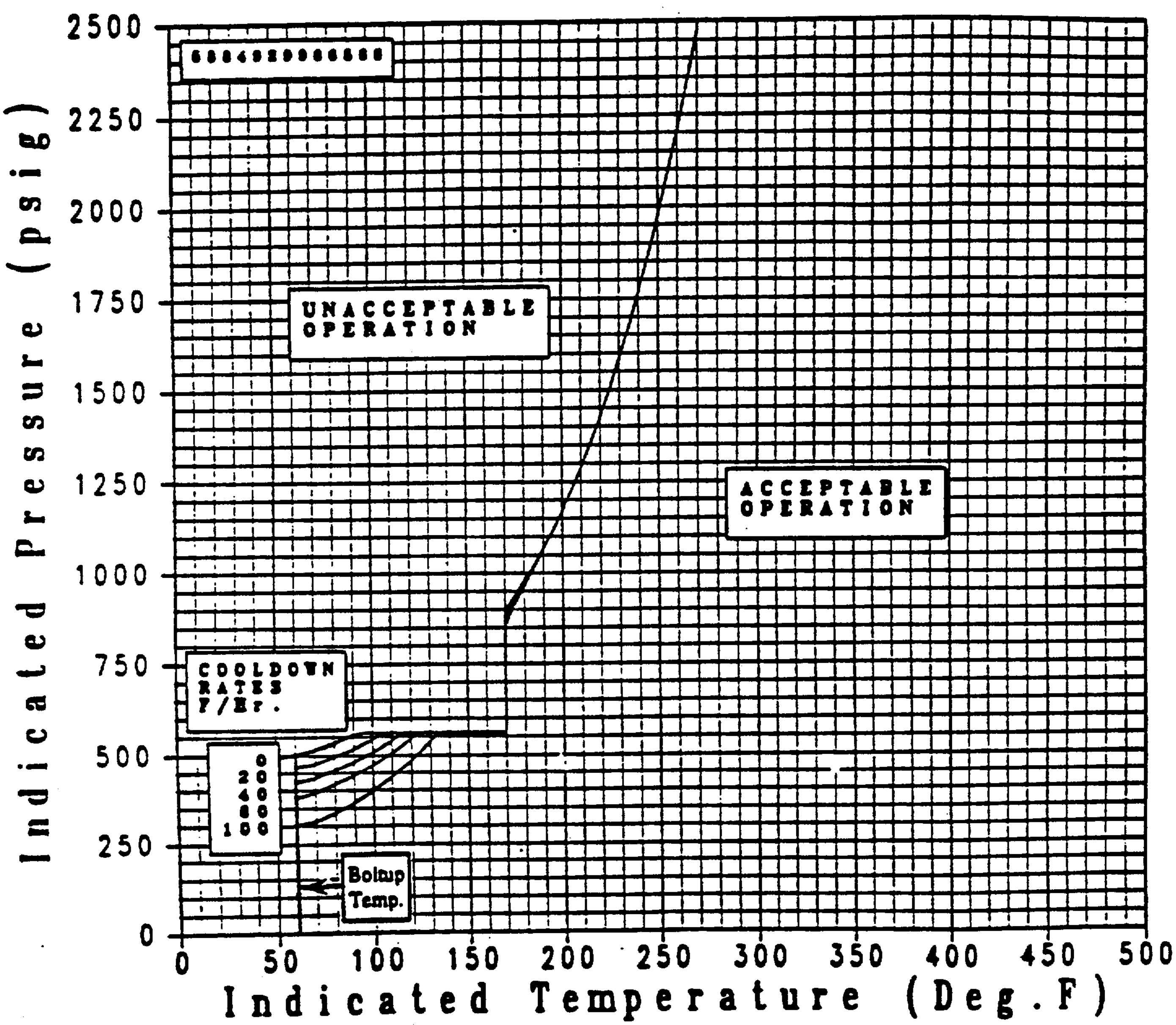


FIGURE 3.4-3

Callaway Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for the First 20 EFPY (With Margins for Instrumentation Errors) Includes Vessel flange requirements of 170°F and 561 psig per 10 CFR 50, Appendix G

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.

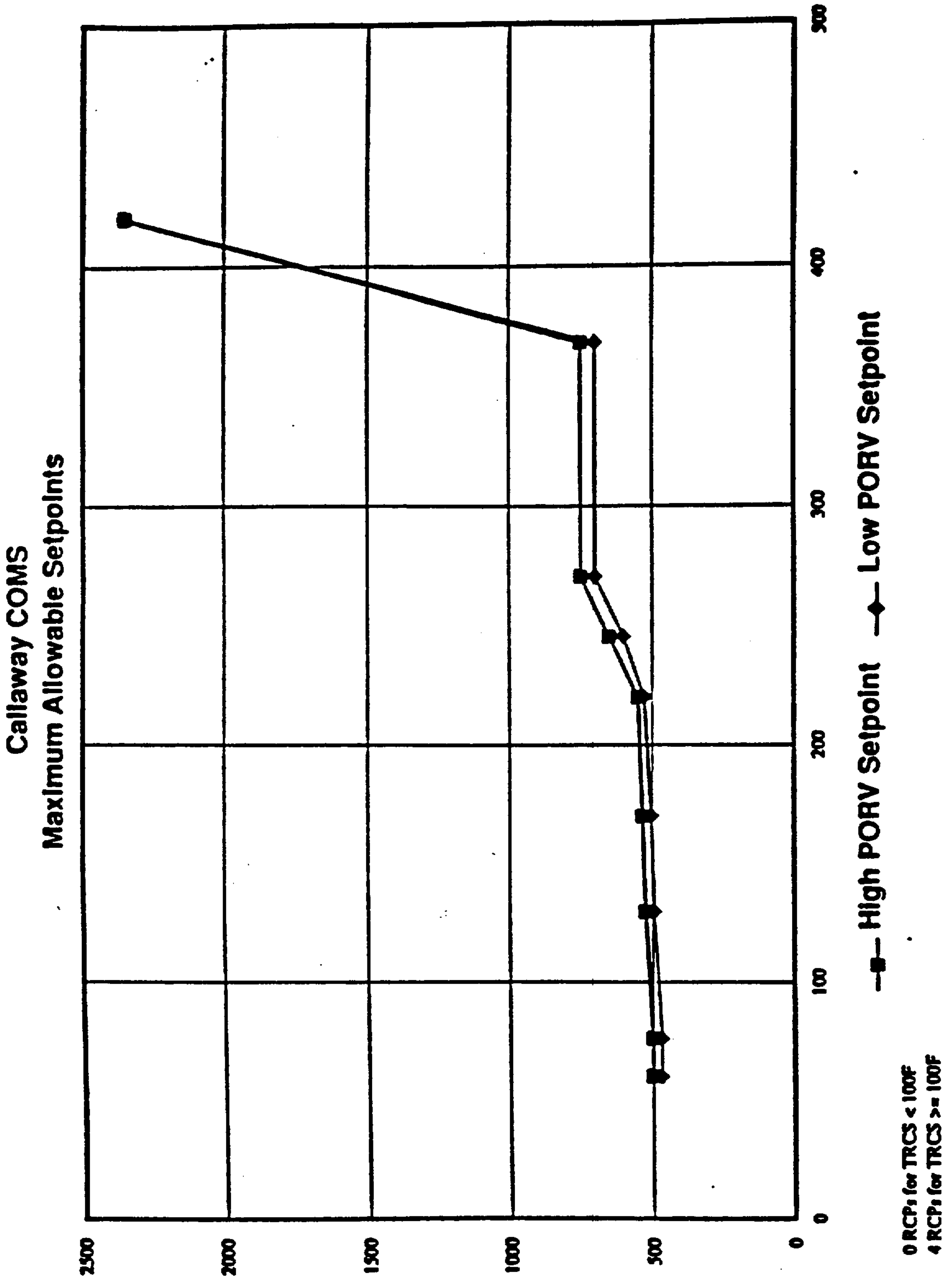
b. For RHR suction relief valve 8708A:

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


Callaway COMS
Maximum Allowable Setpoints
High PORV Setpoint
Low PORV Setpoint
0 RCPs for TRCS < 100F
4 RCPs for TRCS >= 100F
2500
2000
1500
1000
500
0
100
200
300
400
500

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below.

3. System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 20 effective full power years (EFPY) of service life. The 20 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 20 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Capsules are removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The lead factor represents the

PRESSURE/TEMPERATURE LIMITS (Continued)

relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-14894.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, KI, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, 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.

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/or the normal 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 and the normal 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 the normal charging pump and 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 the normal charging pump and 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 the normal charging pump and 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 is updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

ECCS SUBSYSTEMS (Continued)

The limitation for one centrifugal charging pump and the normal 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, which are provided to ensure the OPERABILITY of each component, ensure that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The safety analyses make assumptions with respect to: (1) both the maximum and minimum total system resistance, (2) both the maximum and minimum branch injection line resistance, and (3) the maximum and minimum ranges of potential pump performance. These resistances and ranges of pump performance are used to calculate the maximum and minimum ECCS flows assumed in the safety analyses.

The centrifugal charging pump minimum flow Surveillance Requirement provides the absolute minimum injected flow assumed in the safety analyses. The maximum total system resistance defines the range of minimum flows (including the minimum flow Surveillance Requirement), with respect to pump head, that is assumed in the safety analyses. Therefore, the centrifugal charging pump total system resistance $((P_d - P_{RCS})/Q_d^2)$ must not be greater than 1.004E-02 ft/gpm^2, where P_d is pump discharge pressure in feet, P_{RCS} is RCS pressure in feet, and Q_d is the total pump flow rate in gpm.

The safety injection pump minimum flow Surveillance Requirement provides the absolute minimum injected flow assumed in the safety analyses. The maximum total system resistance defines the range of minimum flows (including the minimum flow Surveillance Requirement), with respect to pump head, that is assumed in the safety analyses. Therefore, the safety injection pump total system resistance $((P_d - P_{RCS})/Q_d^2)$ must not be greater than 0.423E-02 ft/gpm^2, where P_d is pump discharge pressure in feet, P_{RCS} is RCS pressure in feet, and Q_d is the total pump flow rate in gpm.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By letter dated October 17, 1997, as supplemented by letters dated March 3, 1998, and March 17, 1998, Union Electric Company (UE) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. NPF-30) for the Callaway Plant. The proposed changes would revise the Technical Specifications (TS) to modify the plant heatup and cooldown curves and the maximum allowable power operated relief valve setpoint for cold overpressure protection. On March 30, 1998, the staff approved UE's request for an exemption from the requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention for Light Water Nuclear Power Reactors for Normal Operation" in order to apply the American Society of Mechanical Engineers (ASME) Code Case N-514, "Low Temperature Overpressure Protection." The Code case was used in developing the cold overpressure mitigation system setpoints.

The March 3, 1998, and March 17, 1998, supplemental letters provided additional clarifying information that did not change the staff's original no significant hazards consideration determination that was published in the Federal Register on January 14, 1998 (63 FR 2282).

2.0 EVALUATION

2.1 Materials and Fluence

UE's requested amendment is intended to extend the validity of the Callaway Unit 1 P-T limit curves to 20 effective full power years (EFPY). The current P-T limit curves are valid for a service period of 17 EFPY.

The fluence evaluation which is the basis for the proposed revised P-T curves was performed when the third surveillance capsule (V) was removed and evaluated at the end of the eighth cycle. The results are documented in WCAP-14895, "Analysis of Capsule V from the Union Electric Company Callaway Unit 1 Reactor Vessel Radiation Surveillance Program," and includes updates for capsules U and Y which were removed at the end of the first and fourth cycles.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements;" Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," July 12, 1988; GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, March 6, 1992; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2 May 1988; and NUREG-0800, Standard Review Plan (SRP), Section 5.3.2, "Pressure-Temperature Limits." GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2 to review P-T Limit curves. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code, Protection Against Non-ductile Failure."

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G, ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Revision 2 or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2 describes the methodology to be used in calculating the margin term.

2.1.1 Evaluation

For the Callaway Unit 1 reactor vessel, the licensee determined that the most limiting material at the 1/4T and 3/4T locations is the lower shell plate, R2708-3. This plate was fabricated using plate heat C4499-1. The licensee calculated an ART of 100.4°F at the 1/4T location and 84.2°F at the 3/4T location at 20 EFPY. The neutron fluence used in the ART calculation was 7.174 X 10^{18} n/cm^2 at the 1/4T location and 2.547 X 10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting plate was 20°F. The margin term used in calculating the ART for the limiting weld was 34 at the 1/4T location and 32.1 at the 3/4T location, as permitted by Position 1.1 of RG 1.99, Revision 2.

The ART is determined using the chemistry values for each beltline material of Callaway Unit 1. The Reactor Vessel Integrity Database (RVID) contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee provided updated chemistry data for the beltline materials of Callaway Unit 1 by letters dated March 17, 1997 and October 17, 1997 (WCAP-14895). It should be noted that the staff used the updated chemistry values in the review for Callaway Unit 1. In addition, the staff compared the licensee's best estimate chemistry data for weld wire heat 90077 against the best estimate chemistry values in the CEOG Report CE NPSD-1039, Revision 2. The staff verified that the licensee's best estimate Cu and Ni values were the same as the values in the CEOG Report. It should also be noted that the staff is preparing a Request for Additional Information (RAI) about certain aspects of the CEOG Report. In accordance with the RAI, the staff will expect the licensee to address any changes, as needed.

The beltline welds in the Callaway Unit 1 RPV were all fabricated using weld wire heat 90077. The staff reviewed the initial RT_{NDT} values, in the RVID, for welds made of weld wire heat 90077 for all plants. The staff found that the initial RT_{NDT} value of -60°F for the Callaway Unit 1 circumferential and axial welds was acceptable, since there were no other plants with the same weld wire heat.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the Callaway Unit 1 reactor vessel is the lower shell plate, R2708-3, that was fabricated using plate heat C4499-1. The staff's calculated ART value for the limiting material agreed with the licensee's calculated ART value at 20 EFPY. Substituting the ART values for the Callaway Unit 1 limiting plate into the equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 40°F for Callaway Unit 1, provided by the licensee, the staff has determined that the proposed P-T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test.

2.1.2 Conclusion

WCAP-14985 reports the values of the fluence resulting from the measurement of capsule V. In addition, it includes updated values for surveillance capsules U and Y. The update refers primarily to the revision of the cross sections which went into effect after these capsules were evaluated. The approximations used in the analysis of capsule V are the same with those recommended by the staff and are thus acceptable. The calculated and measured values of the fluence (from all the capsules) are in excellent agreement with the corresponding calculated values. The proposed 20 EFPY fluence was derived from these values and due to the good agreement are acceptable.

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 20 EFPY. The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Revision 2 was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the Callaway Unit 1 Technical Specifications.

2.2 Cold Overpressure Mitigation System

2.2.1 Evaluation

UE also requested to modify the plant heatup and cooldown curves and the maximum allowable power operated relief valve (PORV) setpoint curve for cold overpressure protection, as found in TS Figures 3.4-2, 3.4-3, and 3.4-4, respectively. These modifications are necessary for the plant to operate up to 20 effective full power years (EFPY), an increase from 17 EFPYs. In addition, TS Bases 3/4.4.9 and 3/4.5.2 through 3/4.5.4, which stated one of the two centrifugal pumps is allowed by the TS to be operational in MODES 5 and 6 operation with the reactor vessel head installed, are revised by including the "normal" charging pump to the centrifugal charging pump allowed to be operational. The modifications to these TS Bases are acceptable because they provide a consistency with Limiting Condition for Operation (LCO) 3.5.4, which requires all safety injection pumps and one of the two centrifugal charging pumps, but not the "normal" charging pump, to be inoperable while in MODE 5 and MODE 6 operation with the reactor vessel head on.

The cold overpressure mitigation system (COMS) uses PORVs located near the top of the pressurizer to supplement the water relief valves in the residual heat removal system (RHRS) suction lines for protection of the reactor vessel from being exposed to conditions of fast propagating brittle fracture. TS LCO 3.4.9.3 requires that, when the RCS temperature is below 368°F and the RCS is not depressurized with a vent of greater than or equal to 2 square inches, either two RHR suction relief valves, or two PORVs with setpoints not exceeding the limit established in Figure 3.4-4, or one RHR relief valve and one PORV shall be operable. The TS amendment request will revise the PORV setpoints in Figure 3.4-4 to prevent the RCS pressure from exceeding the pressure-temperature limits in the revised Figures 3.4-2 and 3.4-3 for operation up to 20 EFPYs.

The COMS enable temperature and the PORV setpoints for 20 EFPYs are developed using the methodology described in the approved Westinghouse topical report, WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996, as well as the guidelines of ASME Code Case N-514. Code Case N-514 requires the low-temperature overpressure protection (LTOP) systems to be enabled when the reactor coolant temperature is less than 200° F or at temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} (nil-ductility reference temperature) + 50° F, whichever is greater; and to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G. In WCAP-14894, "Callaway Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," July 1997, the RT_{NDT} for 20 EFPYs is calculated to be 100.4° F. Based on the Code Case N-514 guidelines, the COMS enable temperature is 200° F. Therefore, TS LCO 3.4.9.1, which requires the COMS to be enabled at 368°F, is conservative.

The COMS PORV setpoints are determined based on the design basis analyses consisting of a mass input transient and a heat input transient, initiated with the RCS in a water-solid condition and the RHRS isolated from the RCS, disabling the relieving capability of the RHR relief valves. The heat injection scenario is the startup of a reactor coolant pump (RCP) with the steam generator secondary side hotter than the RCS temperature, resulting in a RCS pressurization from sudden heat input to a water-solid RCS from the steam generator. The mass injection scenario is caused by the simultaneous isolation of the RHRS, isolation of letdown and failure of the charging flow controls, resulting in an RCS pressurization from a net charging flow input to a water-solid RCS. The analyses of the heat and mass input transients were performed with the LOFTRAN computer code to determine the RCS pressure overshoot after a PORV is actuated. In addition, the uncertainties associated with pressure and temperature instrumentation, the single failure assumption of a PORV, as well as the hydrostatic and dynamic effects of the RCP are taken into account in the analyses.

In response to the staff's requests for additional information, the licensee in its letters of March 3 and March 17, 1998, provided information related to the analyses performed for development of Callaway COMS setpoints, including input assumptions used in the analyses, the PORV relieving capacity, the injection flow rates of the centrifugal charging pump and the "normal' charging pump. In the heat input transient scenario for the analysis of RCS pressure overshoots, an RCP is assumed to start when the steam generator secondary side temperature is 50°F higher than the RCS cold leg temperature. This 50°F temperature difference assumption is consistent with LCOs 3.4.1.3 and 3.4.1.4.1, which prohibit a RCP from being started in MODE 4 and MODE 5 operation, respectively, unless the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperature. The mass input transient analysis assumes simultaneous injection of both a centrifugal charging pump and the "normal" charging pump into the water-solid RCS while the RHRS and the letdown line are isolated. This assumption is consistent with LCO 3.5.4, which requires all safety injection pumps and one of the two centrifugal charging pump to be inoperable while in MODE 5 and MODE 6 operation with the reactor vessel head on, and therefore, allows a centrifugal charging pump and the "normal" charging pump to be operable under these modes of operation. The combined mass injection rates from both charging pumps at a full capacity over a range of RCS pressures are increased to 105% of the calculated values.

2.2.2 Conclusion

The staff has reviewed the PORV setpoint analyses provided in the UE's March 3 and March 17, 1998, letters, including the RCS pressure overshoots for both the mass and heat input transients for the assumed RCS initial conditions and PORV setpoints, the treatment of pressure and temperature uncertainties, and the selection of the PORV setpoints with comparisons to the Appendix G limits and the limit established for maintaining the integrity of the PORV piping. The results show that, with the selected PORV setpoints at various RCS temperatures, the peak RCS pressures during both the mass and heat input transients are bounded by the Appendix G limits (multiplied by 110% per Code Case N-514 guideline) or the PORV piping pressure limit. Since the analyses were performed with the approved methodology of WCAP-14040, Revision 2, the staff concludes the PORV setpoint curves as specified in the revised Figure 3.4-4 are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Missouri State Official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 2282). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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