September 25, 2000

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

SUBJECT: CALLAWAY PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: PRESSURIZER SAFETY VALVES AND POWER OPERATED RELIEF VALVES (PORVs) (TAC NO. MA9080)

Dear Mr. Randolph:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 25, 2000.

The amendment expands the range of acceptable lift settings and tolerance of the as-found, measured lift settings for the pressurizer safety valves (PSVs) to be considered operable. Following testing, the as-left lift settings of the PSVs would remain at the current tolerance of ±1%, but the nominal lift setting would be reduced. The amendment (1) revises TSs 3.3.2, "Engineered Safety Features Actuation System Instrumentation," 3.4.10, "Pressurizer Safety Valves," and 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," and (2) approves Callaway not having to comply with TMI Action Plan Item II.K.3.1 for automatic PORV block valve closure. The staff has reviewed the changes to the Bases of the TSs and the Callaway Final Safety Analysis Report for the above revisions and has no disagreement with the changes.

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

Sincerely,
/RA/

Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures: 1. Amendment No. 137 to NPF-30
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. 137
License No. NPF-30

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

   A. The application for amendment by Union Electric Company (UE, the licensee) dated May 25, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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. 137 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. This amendment is effective as of its date of issuance and shall be implemented (including issuing the revised EOP E-O and training all the control room operator crews on the revised procedure) before the restart from refueling outage 11, the next refueling outage for Callaway Plant, Unit 1, scheduled to begin in Spring 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

L Raghavan

For Stephen Dembek, Chief, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 25, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 137

FACILITY OPERATING LICENSE NO. NPF-30

DOCKET NO. 50-483

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
3.3-28	3.3-28
3.3-36	3.3-36
3.3-44	3.3-44
3.4-19	3.4-19
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One channel or train inoperable.	F.1	Restore channel or train to OPERABLE status.	48 hours
	OR		
	F.2.1	Be in MODE 3.	54 hours
	AND		
	F.2.2	Be in MODE 4.	60 hours
G. One train inoperable.	---------- NOTE ---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------		
	G.1	Restore train to OPERABLE status.	6 hours
	OR		
	G.2.1	Be in MODE 3.	12 hours
	AND		
	G.2.2	Be in MODE 4.	18 hours
H. One or more trains inoperable.	---------- NOTE ---------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ----------		
	H.1	Declare associated Pressurizer PORV(s) inoperable.	Immediately

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.13	----------NOTE---------- Only applicable to slave relays K602, K622, K624, K630, K740, and K741. ---------- Perform SLAVE RELAY TEST.	18 months AND Prior to entering MODE 4 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days
SR 3.3.2.14	----------NOTE---------- Only applicable to slave relays K620 and K750. ---------- Perform SLAVE RELAY TEST.	18 months AND Prior to entering MODE 3 when in MODE 5 or 6 > 24 hours, if not performed within the previous 92 days

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater (continued)					
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.12	≥ 20.64 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2411 psig and $\leq$ 2509 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 275°F.

---------- NOTE ----------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperature $\leq$ 275°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1% of 2460 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

----------NOTES----------

1. Separate Condition entry is allowed for each PORV.
2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more PORVs inoperable solely due to excessive seat leakage.	A.1	Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable for reasons other than excessive seat leakage.	B.1	Close associated block valve.	1 hour
	AND		
	B.2	Remove power from associated block valve.	1 hour
	AND		
	B.3	Restore PORV to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	----------NOTE---------- Required Actions do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2. ----------		
	C.1	Place associated PORV in manual control.	1 hour
	AND		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable for reasons other than excessive seat leakage.	E.1	Close associated block valves.	1 hour
	AND		
	E.2	Remove power from associated block valves.	1 hour
	AND		
	E.3	Be in MODE 3.	6 hours
	AND		
	E.4	Be in MODE 4.	12 hours

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. NPF-30

UNION ELECTRIC COMPANY

CALLAWAY PLANT, UNIT 1

DOCKET NO. 50-483

1.0 INTRODUCTION

By application dated May 25, 2000, Union Electric Company (the licensee) requested changes to the Technical Specifications (TSs, Appendix A to Facility Operating License No. NPF-30) for the Callaway Plant, Unit 1 (Callaway). The proposed amendment would expand (1) the range of acceptable lift settings for the pressurizer safety valves (PSVs), and (2) the tolerance of the as-found, measured lift settings of tested PSVs, for the PSVs to be considered operable. Following testing, the as-left lift settings of the PSVs would remain at the current tolerance of $\pm 1\%$; however, the nominal lift setting would be reduced.

The amendment would revise TSs 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," 3.4.10, "Pressurizer Safety Valves," and 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," of the Callaway TSs. For TS 3.3.2, a new Action H for one or more trains inoperable would be added, the note for surveillance requirement (SR) 3.3.2.14 would be revised to identify another slave relay that the SR would be applicable to, and the automatic PORV actuation would be added to Table 3.3.2-1, "Engineered Safety Features Actuation System Instrumentation." For TS 3.4.10, the range of allowable PSV lift settings in the limiting condition for operation (LCO) would be expanded from $\geq$ 2460 and $\leq$ 2510 to $\geq$ 2411 and $\leq$ 2509, and SR 3.4.10.1 would be revised to state that, following testing, the lift settings shall be "within 1% of 2460 psig" instead of simply "within 1%." The nominal PSV lift setting would be changed from 2485 psig to 2460 psig, because the maximum PORV lift setting would not be increased and the minimum setting would be reduced 59 psig. For TS 3.4.11, Actions A and B would be revised to be actions for inoperable PORVs either solely due to excessive PORV seat leakage (Action A) or for reasons other than excessive seat leakage (Action B), and Action E would remain an action for two inoperable PORVs, but would be only for reasons other than excessive seat leakage.

The licensee also provided revisions to the Bases of the TSs and the Callaway Final Safety Analysis Report (FSAR) for the above changes, in Attachments 4 and 5 of the May 25, 2000, application, respectively.

2.0 BACKGROUND

The current allowable PSV lift settings (i.e., the pressurizer pressure at which the valves open) in LCO 3.4.10 are $\geq$ 2460 and $\leq$ 2510 psig. This is a nominal setting of 2485 psig, with a $\pm$ 1% tolerance, for the PSVs to be considered operable. The licensee has proposed to lower the nominal lift setting by 1% and increase the setting tolerance for valve operability by 1%; therefore, the proposed allowable lift settings are $\geq$ 2411 psig and $\leq$ 2509 psig (i.e., a nominal 2460 psig $\pm$ 2%). Thus, the upper limit of the allowable PSV lift setting would not be increased by the proposed changes; in fact, it would be lowered by 1 psig to 2509 psig. Also, the tolerance for resetting the PSV lift setting after testing, if needed, will remain unchanged at $\pm$ 1% (as stated in SR 3.4.10.1). In revising the lift settings of the PSVs, the licensee has proposed changes to TS 3.4.10, "Pressurizer Safety Valves."

To prevent water passing through the PSVs in an inadvertent-emergency core cooling system (ECCS)-actuation-at-power event, the licensee will modify and upgrade the PORV actuation circuitry to fully Class 1E (some components are already Class 1E) to take credit for automatic action of at least one PORV during the event. The modification is addressed in Attachment 6 to the application. The reanalysis of the inadvertent-ECCS-actuation-at-power event credits operator actions within the control room during the event to terminate the normal charging pump (NCP) flow, to open a PORV block valve, and to assure that the PORV handswitches are in the automatic operation position.

In upgrading the PORV actuation circuitry and taking credit for automatic actuation of at least one PORV, the licensee has also proposed changes to TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," and TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)."

The requirements of the Three Mile Island (TMI) Action Plan in NUREG-0737 (dated November 1980) include Item II.K.3.1, "Installation and Testing of Automatic Power-Operated Relief Valve Isolation System." All pressurized water reactor (PWR) licensees, including that for Callaway, had to provide a system that uses the PORV block valve to protect against a small-break loss-of-coolant accident (SBLOCA). This system would automatically cause the block valve to close when the reactor coolant system (RCS) pressure decays after the PORV has opened. The licensee stated in its application that the Callaway design, as described in FSAR Section 18.2.17.1.2, meets Action Plan Item II.K.3.1 in that the PORV block valve design includes the capability to automatically isolate the PORVs.

The licensee is now proposing to remove the capability for the PORV block valves to automatically isolate the PORVs. This is based on Westinghouse Topical Report, WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants," dated February 1981. This topical report was reviewed by the staff for Indian Point Nuclear Generating Station, Unit No. 2 (Docket No. 50-247) and accepted as the basis to conclude that an automatic PORV block valve isolation system was not required for Indian Point Unit 2. The licensee has requested that the staff apply this topical report to Callaway and approve that an automatic PORV block valve isolation system is also not required for Callaway. In the conference call of June 28, 2000, the licensee stated that WCAP-9804 remains applicable to Callaway.

3.0 EVALUATION

There are three PSVs and two PORVs on the pressurizer. The PORVs are to prevent challenges to the PSVs, including water flow through the PSVs, during non-accident conditions. The PSVs provide overpressure protection of the RCS and the upper lift setting limit is based on having only the PSVs to limit the RCS pressure surge at full power to 110 percent of the RCS design pressure. For conservatism, no credit is taken for the two PORVs.

Accident Analyses

The maximum allowable PSV lift setting is set by the locked reactor coolant pump rotor and loss of external electrical load/turbine trip accidents for overpressure protection. The licensee stated that all other non-loss-of-coolant accidents that take credit for the maximum allowable PSV actuation are less sensitive to the assumed setting and are not adversely affected by the proposed amendment. Because the proposed change to the PSV lift settings does not increase the maximum PSV lift setting (in fact the maximum setting is decreased by 1 psig), upon which the current analyses on RCS overpressure protection are based, the proposed amendment does not affect these analyses, and the consequences of these accidents remain conservative and bounding.

The limiting minimum allowable PSV lift setting is based on the inadvertent-ECCS-actuation-at-power event. The minimum setting establishes the time requirements for operator actions to prevent the PSVs opening with the pressurizer water solid. A lower PSV setting would require earlier operator actions to preclude water passing through the PSVs.

A reanalysis of the inadvertent-ECCS-actuation-at-power event was performed for Callaway. The current analysis credits operator termination of safety injection within 10 minutes. The licensee stated that this action within 10 minutes is questionable under the current version of emergency operating procedure (EOP) E-O, "Reactor Trip or Safety Injection" at Callaway. The reanalysis credits the following operator actions in the control room during the event: terminate NCP flow at six minutes into the event, and open the PORV block valve, because there is a block valve downstream of each PORV, and assure the PORV handswitches are in the automatic operation position at nine minutes into the event. The licensee stated that the above operator action times include all process and instrumentation delays, and the stated times will prevent the situation where the PSV lift setting is challenged (i.e., a PSV lifts open) and water would enter the opened PSV. Instead, a PORV would automatically open and relieve the pressure in the RCS to prevent the pressure from rising to the lower lift setting of a PSV.

The licensee has proposed to allow the PORVs to automatically relieve pressure (preventing the PSVs from opening) instead of terminating the ECCS flow to prevent the RCS pressure rise (and the PSVs from opening) because the ECCS flow may be necessary and, until it is determined that the flow is not necessary, the ECCS should be allowed to continue. The licensee has decided to relieve pressure through the PORVs instead of terminating the ECCS flow, as is done in the current analysis for the inadvertent-ECCS-actuation-at-power event. If the licensee had elected to continue the current analysis for the event, it would have to terminate ECCS flow in a shorter time than the current 10 minutes.

The smallest difference in the lift settings between a PSV and PORV would be the difference between the lowest setting for a PSV (i.e., the proposed 2411 psig, which is rounded up from 2460 psig minus the 2% tolerance, or 2410.8 psig) and the 2335 psig for a PORV (from Table 5.4-11 of the FSAR). The unchanged PORV setting is at least 75 psi below the proposed new lower setting of the PSVs. Therefore, the proposed amendment is a reduction in the difference between the PORV-PSV actuation settings from 125 psi to 75 psi; however, the difference of 75 psi is considered sufficient so that the PORV would actuate before a PSV in the case of rising RCS pressure.

To verify the operator action times given above to allow for automatic actuation of the PORV before any actuation of a PSV, the licensee conducted simulator exercises of the inadvertent-ECCS-actuation-at-power event on August 10, 1999, to determine the times required for the control room operators to stop the NCP, unblock the PORV, place the PORV handswitches in the automatic operation position, and terminate the ECCS flow. The licensee stated in its application that in all cases the NCP was stopped within 4 minutes, the PORVs were unblocked and available for automatic operation within 7 minutes, and ECCS flow was terminated within 21 minutes. The reanalysis for this event is incorporated in the draft FSAR changes shown in Attachment 5 to the application and takes credit for the longer times of 6 minutes to stop the NCP and 9 minutes to have the PORVs capable for automatic operation. It is not necessary for the licensee to terminate the ECCS flow within the current analysis of operator action within 10 minutes to terminate safety injection, during the inadvertent-ECCS-actuation-at-power event, because the PORVs are available to automatically relieve RCS pressure.

The simulator exercises were performed by two control room operator crews that were minimally staffed with three licensed operators. To support the above operator action times in the reanalysis of the inadvertent-ECCS-actuation-at-power event, the licensee explained that EOP E-O was revised and the simulator exercises were performed using the revised procedure. The two crews were aware of the exercise. The revised procedure will be issued during the implementation of the amendment and all operating crews will be trained on the revised E-O procedure during their scheduled requalification training. These results provide the staff with reasonable assurance that the operators can perform the required actions within the newly established 9-minute time limit. In the conference call of June 28, 2000, the licensee stated that the actions required by the control room operators are taken within the control room and the requalification training will be completed for all the control room operator crews before the amendment is implemented in the April 2001 refueling outage.

If the operators should take more than the 9 minutes, water could pass through a PSV and the PSV could be damaged such that the valve would not seat properly, and there could then be uncontrolled leakage from the RCS (i.e., the leakage could not be stopped while the plant was at power). Because temperature is measured in the safety valve discharge line, the licensee will have indication of the PSV leakage from an increase in the temperature and the RCS operational leakage limits in LCO 3.4.13 will require the licensee to shut down the plant if the identified leakage (i.e., the leakage through a safety valve) is greater than 10 gpm for more than 4 hours. This leakage is accounted for and analyzed in the Callaway design and discussed in FSAR Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety or Relief Valve." The results of the analysis show that the pressurizer low pressure (from the decreasing pressure following the open PSV) and overtemperature deltaT reactor protection signals

provide adequate protection against this RCS depressurization event, and the TSs will prevent the licensee from operating with a high leak (> 10 gpm) through the safety valves. However, before the pressurizer pressure reached the proposed expanded PSV pressure settings, the pressure would go above the high pressurizer pressure reactor trip settings (> 2393 psig in Table 3.3.1-1 of the TS) and there should be a reactor trip. In addition, the licensee stated that the consequences of failure of the operators to open the PORV block valve (which would prevent the PORVs from relieving rising RCS pressure) would be bounded by the small break loss-of-coolant accident (LOCA) and Callaway has been designed for the small-break LOCA. The staff agrees with the licensee's statement.

The licensee stated that the design basis large break LOCA, steam generator tube rupture, and main steamline break analyses all result in decreasing RCS pressure and, therefore, will not challenge the PSVs. Therefore, these analyses are not affected by the proposed amendment.

The proposed amendment does not affect any reactor protection or engineered safety features system trip settings. Also, the licensee stated that (1) the pressurizer surge line layout design calculation, and (2) the pressurizer relief tank (PRT) level alarm setpoint calculation use the nominal PSV lift setting as an input and, because the nominal setpoint is being reduced in the proposed amendment, the reduced value is conservative and, therefore, will not affect these calculations. The PRT must have adequate volume for 110% of the steam volume from the pressurizer and reducing the lift settings in the proposed amendment will reduce the steam volume released from the pressurizer to the PRT.

Based on the above, the staff concludes that the proposed PSV nominal setting of 2460 psig $\pm$ 2% psig is acceptable. After the valves are tested, the settings will reset if they are not within 2460 psig $\pm$ 1% psig. The $\pm$ 1% tolerance for the as-left settings is not being changed by the proposed amendment; therefore, there will be less chance that the drift of the as-left setting will exceed the upper setting limit used in the overpressure analysis. The licensee also stated that the current design transient limits for the RCS assume a $\pm$ 3% tolerance on the PSVs so the proposed amendment of a $\pm$ 2% tolerance does not affect the design transient limits for the RCS. Based on this, the staff concludes that the proposed changes to LCO 3.4.10 to have the PSV lift settings $\geq$ 2411 psig and $\leq$ 2509 psig (i.e., 2460 psig $\pm$ 2%) is acceptable. Because the addition of the phrase "of 2460 psig" to SR 3.4.10.1 will clarify what pressure the $\pm$1% applies to in the SR, the proposed revision to the SR is acceptable.

Changes to PORV Automatic Actuation Logic

The current PORV automatic actuation logic is not Class 1E. Because the licensee will be taking credit for this logic in the inadvertent-ECCS-actuation-at-power event, it has decided to upgrade the logic to be fully Class 1E. The staff agrees that this is necessary to take credit for the automatic actuation of the PORVs during this event.

In its application, the licensee explained the current opening logic is control grade circuitry that actuates on a one-out-of-one (1/1) logic and the current closure logic is safety-grade based on a (2/4) pressurizer pressure low logic. The licensee explained that to upgrade the automatic actuation logic to fully Class 1E, the closure logic will be used in the design of the new safety grade opening logic. The new (2/4) opening logic will open the PORVs and the new (3/4)

closure logic will close the valve when the pressure drops 20 psi below the opening pressure of 2335 psig.

The licensee stated that the new design will minimize the potential for spurious openings of the PORVs; however, two failed high channels of pressurizer pressure will result in an inadvertent opening of both PORVs and the PORVs would remain open until they were remote manually closed. With the current control logic, while a single channel failing high would open one PORV, the closure logic would close that PORV when the pressurizer pressure dropped below about 2200 psia. The current design would require 3 channels failing high for both PORVs to open and remain open. Therefore, the licensee concluded that the new logic increased the probability of an inadvertent opening of the PORVs and them remaining open. However, the licensee concluded that the new logic will add assurance that the PORVs can provide automatic pressure control and prevent challenges to the PSVs when the pressurizer is water solid, the higher probability is from multiple failures which are highly unlikely, and the operators can act to close the PORVs from the control room. Therefore, the licensee concluded that the increased probability was negligible and the new control logic was beneficial to the safety design of Callaway.

The licensee has provided the revised FSAR pages for the changes in PORV control logic and the revised PORV control schematics in Attachments 5 and 6 to the May 25, 2000, application. The staff has reviewed these attachments and does not have any disagreement with the licensee. The licensee has also proposed changes to TS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," to include the upgraded PORV automatic actuation logic in the TS because the licensee will now be taking credit for this logic in the inadvertent-ECCS-actuation-at-power event.

The licensee has proposed to add the automatic pressurizer PORV actuation to Table 3.3.2-1, "Engineered Safety Features Actuation System Instrumentation." This will add the applicable reactor modes, required channels, action conditions, surveillance requirements, and allowable value for the control logic, as is true for other ESFAS instrumentation taken credit for in accident analyses. The licensee is also (1) adding an Action (or Condition) H for the PORV actuation instrumentation being inoperable, and (2) revising the note to SR 3.3.2.14 for performing slave relay tests to add another slave relay that the SR is applicable to (i.e., slave relay K750 is being added to the SR).

The licensee has proposed to add the automatic actuation logic and actuation relays and the pressurizer pressure high to Table 3.3.2-1 for the automatic PORV actuation logic. The pressurizer pressure high is the input signal to the actuation logic. The proposed applicable modes for the PORV actuation are the same as the applicable modes for the operability of PORVs in TS 3.4.11. The licensee stated there are two trains for controlling the PORV and there are four channels for the pressurizer pressure high signal. The SRs proposed for the PORV actuation logic and input signal are consistent with other automatic actuation logic and actuation relays, and input signals for Table 3.3.2-1. There are no allowable values for the PORV automatic actuation logic and actuation relays, and the allowable value for the input signal is the lift setting in psia for the PORVs.

The proposed Action H will have the note that one train may be bypassed for up to four hours for surveillance testing provided that the other train is operable. The proposed note is the same note as in Actions C and G in TS 3.3.2 for two train automatic actuation logic and actuation relays in Table 3.3.2-1 that are similar to the PORV automatic actuation logic and actuation relays, and is acceptable. The four hours is based on reliability analysis and the average time required to perform the channel surveillance. However, the proposed required action for Action H is not the same as that in Actions C and G. The proposed required action, as explained in Attachment 4 to the application for revised TS Bases pages, addresses the ability to mitigate an inadvertent-ECCS-actuation-at-power event with one or more automatic actuation logic trains inoperable. The licensee stated that the appropriate action is to declare the associated PORV inoperable immediately and enter the actions in TS 3.4.11 for the PORVs. The staff agrees with the licensee's conclusion.

For existing SR 3.3.2.14, which requires testing of slave relays, the licensee has proposed to add the relay K750 by revising the note for the SR. The licensee is stating that adding the PORV automatic actuation logic and actuation relays to Table 3.3.2-1 is also adding slave relay K750 to SR 3.3.2.14. This SR is listed in the applicable SRs for this logic in the table. The licensee stated that the 18-month frequency of the surveillance was accepted by the staff at the licensing of Callaway and is consistent with the frequency in SR 3.4.11.2 for performing a complete cycle of each PORV in accordance with the inservice testing program. The 18-month frequency is consistent with relief request BB-10 on the PORVs, which was approved in the staff's letter of March 19, 1999 (TAC No. MA4469). The K750 slave relay must be tested at cold shutdown because to test the relay would open the PORV and depressurize the RCS because the block valves must be open for the test.

Based on the above, the staff concludes that the proposed changes to TS 3.3.2 are acceptable.

Changes to PORV Block Valve Automatic Actuation Logic

There is a PORV block valve downstream of each PORV. The licensee stated that the current safety grade PORV closure logic is also used to close the block valves. In the main control room, the block valve handswitches on the main control board can be in any of the "OPEN," "AUTO," or "CLOSE" positions. Currently, in the AUTO position, the 2/4 closure logic automatically closes the block valves when the pressurizer pressure drops below about 2185 psig. For the PORV automatic open logic upgrade discussed above, the safety grade PORV/PORV block valve closure logic will be used in the safety grade PORV automatic opening logic and, because the block valves are not designed for continuous cycling, the licensee stated that it is undesirable for the block valves to automatically open and close with the PORVs during the inadvertent-ECCS-actuation-at-power event. Therefore, the AUTO feature for the block valves is to be eliminated by the licensee and, after the modification is implemented, the block valves are to be manually opened and closed from the main control room. The block valve handswitches will be changed from "OPEN-AUTO-CLOSE" to "OPEN-CLOSE."

In TMI Action Item II.K.3.1 of NUREG-0737, the Commission required automatic closure of the block valves to prevent a small break LOCA resulting from a stuck open PORV. As stated in FSAR Section 18.2.17.1.2, the Callaway design when it was licensed included the capability to automatically close the PORV block valves if the PORVs fail to close and the RCS pressure

drops below the reset pressure of the PORVs. This is described in FSAR Sections 7.6.6 and 7.6.10. However, the licensee stated in Section 18.2.17.1.2 that the Westinghouse Owners Group evaluated the need of having such an automatic PORV isolation system and this evaluation is documented in Westinghouse Topical Report WCAP-9804. The licensee stated that it has concluded that the automatic PORV isolation system at Callaway should not be required at Callaway. The licensee proposed to eliminate this system as part of its proposal to expand the PSV lift settings down to 2411 psig and install the new PORV and PORV block valve actuation logic discussed above.

Westinghouse Topical Report WCAP-9804

The staff evaluated WCAP-9804 in a review for the Indian Point Unit 2 plant for which its licensee also requested that an automatic PORV isolation system not be required. The staff issued its evaluation of WCAP-9804 and concluded that the isolation system was not required in its letter of September 13, 1983 for Docket No. 50-247. The following evaluation of WCAP-9804 is condensed from the safety evaluation (which included a Technical Evaluation Report from the staff's contractor) attached to the letter.

The Westinghouse report WCAP-9804 considered a spectrum of initiating events that lead to PORV/SV opening. The report referred to the PSVs as simply safety valves (SVs) and the acronym SV will be used hereafter. The event tree methodology was used to determine various possible outcomes due to the initiating events and to estimate the SBLOCA frequencies due to a stuck-open PORV/SV (i.e., the SBLOCA-PORV/SV frequencies).

The initiating event frequencies were based on the generic estimates for PWRs given in Electric Power Research Institute (EPRI) NP-801, "ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients," dated July 1978, and the estimates of SBLOCA-PORV/SV frequencies were obtained from the frequencies of the initiating events, the probabilities of exceeding PORV/SV setpoints given the initiating events, the availabilities of the PORV block valves, the PORV/SV failure probabilities, and the probability of operator error.

In addition, the WCAP-9804 considered the impact of post-TMI modifications on probability data and compared pre-TMI results with post-TMI results. Finally, Westinghouse performed sensitivity analyses on post-TMI results to assess the impact of the following parameters:

- safety injection system pressure difference (high-head vs. low-head),
- probability of PORVs being blocked off, and
- probability of operator error.

Based on the staff's review of WCAP-9804, the staff finds that the event tree methodology used in the report is a valid approach to estimating the SBLOCA-PORV/SV frequencies. The staff finds that most of the probabilistic data in the event tree appear reasonable with a few exceptions, for example, the PORV/SV failure probabilities. The Westinghouse analysis also includes a few stuck-open PORV/SV scenarios due to a spurious actuation of a safety injection system.

The results of the Westinghouse analysis (with credit for operator action) indicate that the post-TMI SBLOCA-PORV frequency is about 2×10^{-6}/reactor-year for a plant with a high-head

safety injection system and is about 10^{-6}/reactor-year for a plant with a low-head system. In addition, the post-TMI SBLOCA-SV frequency is about $5x10^{-6}$/reactor-year for a plant with a high-head system. As discussed in the attached TER, the contractor has performed calculations and verified these estimates, using the Westinghouse data. However, the staff believes that the following considerations should be incorporated in the Westinghouse analysis:

- PORV/SV Failure Probability: Westinghouse used 10^{-3}/demand as the PORV/SV failure probability to reset. This is the failure probability per opening, not per transient. The staff believes that the PORV failure probability may be an order of magnitude higher if the PORV failure event at Ginna and the PORV failure event at North Anna 2 (Licensee Event Report 80-29) are also included in estimating the PORV failure probability. Therefore, the SBLOCA-PORV frequency may increase by an order of magnitude.

- PORV Block Valve Availability: The Westinghouse analysis assumes that 45% of the time PORVs are not blocked off. If a plant operates with PORVs not blocked off all the time, the SBLOCA-PORV frequency may increase by about a factor of two. By the same token, if a plant operates with PORVs blocked off all the time, the SBLOCA-SV frequency may also increase by a factor of two.

- Multiple PORV Openings: The Westinghouse calculation of SBLOCA-PORV frequency assumes a PORV opens once per transient. Most Westinghouse plants have two PORVs, and a few have three PORVs. Therefore, depending on the load rejection capabilities, it is not uncommon for a PORV to open several times or for multiple PORVs to open during an overpressure transient. Therefore, the staff believes that the effect of multiple PORV openings should be included in estimating the SBLOCA-PORV frequency.

The staff also did a review to ascertain whether the generic topical report applies to a specific Westinghouse plant. The staff reviewed plant-specific information such as PORV/SV challenge frequencies, the fraction of time the block valves are closed, and the various post-TMI modifications that may have reduced the PORV/SV challenge frequencies. Based on this review, the staff believes WCAP-9804 is applicable to all plants listed in Table 1 of the safety evaluation in the staff's letter of September 13, 1983, except McGuire Unit 1.

The staff reviewed the PORV/SV failure and challenge data for April 1, 1980, to March 31, 1983, from the reports submitted by PWRs in response to TMI Action Plan Item II.K.3.3, "Reporting SV and RV Failures and Challenges," which required PWR licensees to periodically report such challenges beginning April 1, 1980. In the safety evaluation dated September 13, 1983, the staff stated that there were no PORV/SV challenges during the 3-year period for many of the Westinghouse plants listed in Table 1, and the maximum number for the 3-year period was 4 challenges. Using 4 PORV challenges in 3 years, the staff estimated that the upper 95% confidence limit on the PORV challenge frequency is about 3.1/reactor year. And assuming that (1) the PORVs are not isolated, (2) the PORV failure probability is 10^{-2}/demand, and (3) the operator probability in not isolating the stuck-open PORV is $5x10^{-2}$/demand, the staff estimated that the SBLOCA-PORV frequency is about $5x10^{-3}$/reactor year, which still remains within the range of the SBLOCA frequency given in WASH-1400, "Reactor Safety Study," dated October 1975, the first risk assessment of nuclear power plants (i.e., 10^{-2} to $^{-4}$/reactor-year). The SBLOCA-PORV frequency is conservative because the operational data of 4 challenges in

3 years is bounding for the Westinghouse PWRs and the 95% confidence limit is used. Moreover, depending on the fraction of time that PORVs are blocked off, the challenge frequency may be less. For example, plant-specific data indicates that, for about 33% of the time during the 1980 to 1983 period, PORVs have been blocked off. The staff estimated that the SBLOCA-PORV frequency would be about 10^{-2}/reactor-year, considering the fraction of time the PORV block valves were closed for all plants in Table 1.

The staff also modified the Westinghouse estimate of the SBLOCA-SV frequency with the following assumptions: there are in general 3 SVs in a Westinghouse plant, PORVs are blocked off all the time due to leakage, the SV failure probability is 10^{-2}/demand according to WASH-1400 and NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One, Unit 1 Nuclear Power Plant," dated June 1982. The staff estimated that the SBLOCA-SV frequency is about $3x10^{-4}$/reactor-year, which falls toward the lower end of the range of the SBLOCA frequency given in WASH-1400 (i.e., 10^{-2} to $^{-4}$/reactor-year).

The staff estimate was obtained as follows. Westinghouse estimated the SBLOCA-SV frequency as $5x10^{-6}$/reactor-year; however, their challenge frequency did not take into account that all three SVs may be challenged. Moreover, the Westinghouse analysis assumed that PORVs are blocked about 50% of the time. If PORVs are blocked, the probability of a SV lifting is about 10 times as high as if the PORVs are not blocked, according to Table 3-4 of WCAP-9804. Hence, for a plant with PORVs blocked all the time, it is appropriate to double the SV challenge frequency used by Westinghouse. In addition, Westinghouse assumed that the probability of a SV failing to close is 10^{-3}/demand, while the staff estimated this probability as 10^{-2}/demand. The net result is that the Westinghouse estimate of the SBLOCA-SV frequency should be multiplied by a factor of 3 because of a possibility that all 3 SVs are challenged, by a factor of 2 because the PORVs at a plant may be blocked off all the time, and by a factor of 10 because of the staff's estimate of the probability of a SV failing to close per demand. This leads to an overall increase by a factor of 60 and to the staff's SBLOCA-SV frequency of $3x10^{-4}$ /reactor-year (i.e., 60 x $5x10^{-6}$/reactor-year). This frequency agrees with the data that there have been no challenges to SVs in Westinghouse plants in over 200 reactor-years of operation.

The staff also addressed PORV leakage in the safety evaluation dated September 13, 1983. The staff stated that WCAP-9804 stated that PORVs in Westinghouse plants are blocked off about 55% of the time. The intentional blocking of PORVs is done to eliminate RCS leakage through PORVs to ensure that the leakage does not exceed the limit in the TSs. Because there are many Westinghouse plants that have blocked-off PORVs, it may imply either that PORVs need to be modified to correct the leakage problem or that there should be some maintenance or repair work on PORVs on a periodic basis. A plant that operates with blocked-off PORVs will depend on the SVs to relieve RCS overpressure. Considering the fact that the SV capacity is much larger than the PORV capacity and there is no block valve to terminate a SV release, the consequences of a stuck-open SV will be more severe than that for a stuck-open PORV. In addition, if PORVs are not blocked off, they supply additional pressure relieving capacity to the RCS in an anticipated transient without scram (ATWS) event. It would appear prudent to limit the time that plants operate with PORVs blocked off. Although the staff stated in the safety evaluation that it was considering the need to have a limit in the TSs on the amount of time a plant can operate with PORVs blocked, it has since been decided that this limit is not necessary.

The staff concluded in its safety evaluation dated September 13, 1983, based on its review of WCAP-9804 and its estimate of SBLOCA-PORV/SV frequencies, that the SBLOCA-PORV/SV frequencies were low enough that an automatic PORV isolation system was not required for Indian Point Unit 2.

In the conference call of June 28, 2000, the licensee stated that the plant has not been operated with PORVs blocked since about 1985/1986 time frame, and there have only been two challenges to the PSVs, which occurred during reactor trips as expected from the plant design. The PORVs were modified after the 1985/1986 time frame to solve leakage problems. Based on this plant-specific PORV/SV data, the staff concludes that WCAP-9804 still applies to Callaway and the staff estimates for SBLOCA-PORV/SV discussed above are conservative for Callaway.

The current analysis for the inadvertent-ECCS-actuation-at-power event is that operator action will terminate safety injection within 10 minutes to prevent the pressurizer going water solid and that water passes through a PSV. The reason the licensee has proposed to use operator actions during the event to have the PORVs automatically open instead of turning off safety injection is to prevent water from entering the PSVs. Based on its review of WCAP-9804 and estimate of SBLOCA-PORV/SV frequencies, and on the Callaway plant-specific date, the staff also concludes that SBLOCA-PORV/SV frequencies are low enough that a PORV isolation system is not required at Callaway.

Changes to PORV TS 3.4.11

The licensee has proposed to revise Action Conditions A, B, and E of TS 3.4.11, "Pressurizer PORVs." The licensee has proposed to make a distinction between PORV inoperability due to (1) excessive seat leakage, or (2) reasons other than excessive seat leakage. For these two cases, the former would be the same as the case in the actions of the PORVs being able to be manually cycled and, therefore, could be used to relieve RCS pressure and prevent an PSV from having to open, with rising RCS pressure. And the latter would be the same as the case in the actions of the PORVs not being able to be manually cycled and, therefore, could not be used to relieve RCS pressure and prevent a PSV from having to open with rising RCS pressure.

Therefore, the licensee is proposing to replace (1) the phrase "and [PORVs] capable of being manually cycled," in Conditions A and E with the phrase "solely due to excessive seat leakage," and (2) the phrase "and [PORVs] not capable of being manually cycled" in Condition B with the phrase "for reasons other than seat leakage." The proposed change in the phrases does not change the required actions or the allowable operating times for the action conditions.

Because the change in the phrases does not change the meaning of the phrases, but only changes the wording to clarify why the PORVs are inoperable and can be manually cycled, the staff concludes that the proposed changes to the action conditions of TS 3.4.11 for PORVs is acceptable.

Conclusion and Implementation of the Amendment

Therefore, based on the above evaluation, the staff concludes that the proposed amendment is acceptable. In its application, the licensee stated that the amendment was to be implemented before the plant restart from refueling outage 11, which is the refueling outage scheduled to begin April 2001. The implementation date is based on the PSVs having to be removed from the plant during this outage to have the lift settings changed to meet the settings approved in this amendment, issuing the revised EOP E-O, and completion of the training of all the control room operator crews on the inadvertent-ECCS-actuation-at-power event with the revised EOP E-O.

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 (65 FR 39962). 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.

Attachment: Technical Evaluation Report by Franklin Research Center

Principal Contributor: Jack Donohew

Date: September 25, 2000