Union Electric
Callaway Plant

PO Box 620
Fulton, MO 65251

May 1, 2000

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

ULNRC-4232

Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSEE EVENT REPORT 1999-008-01
Reactor Trip Due To Low Steam Generator Water Level
Resulting From Loss Of Power To Feedwater Control Cabinet
(Reference: ULNRC-4166, dated December 22, 1999)

The enclosed licensee event report is submitted in accordance with 10CFR50.73(a)(2)(iv) to report an event that resulted in the automatic actuation of an Engineered Safety Feature and the automatic actuation of the Reactor Protection System. This revised licensee event report is submitted in order to clarify wording within the safety significance section which stated that no radioactive materials were released to the environment during this event. Although no unanticipated quantities of radioactive material were released, insignificant amounts of radioactive material normally present within the secondary coolant were released via the Atmospheric Steam Dumps and the Turbine Driven Auxiliary Feedwater Pump steam exhaust during this event.

R. D. Affolter
Manager, Callaway Plant

RDA/mdhu

Enclosure

IE22

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LICENSEE EVENT REPORT (LER)

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TITLE (4) **REACTOR TRIP DUE TO LOW STEAM GENERATOR WATER LEVEL RESULTING FROM LOSS OF POWER TO FEEDWATER CONTROL CABINET**

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 1	2 6	9 9	9 9	- 0 0 8 -	0 1	0 5	0 1	0 0		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : *(Check one or more of the following)* (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	X 50.73(a)(2)(iv)	OTHER *(Specify in Abstract below or in Text, NRC Form 366A)*
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (AREA CODE)
J. D. Schnack, Supervising Engineer, QA Regulatory Support	5 7 3 6 7 6 - 4 3 1 9

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	J B	J X	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines)(16)

On 11/26/99 at 1227 CST with the plant at 100% power, a reactor trip occurred on low steam generator level due to a momentary power loss in control cabinet RP043. Annunciators initially alerted the control room of a power failure in this cabinet. Investigations revealed the primary power supply was deenergized and the secondary power supply was supplying the cabinet. An Equipment Operator (EO) was dispatched to investigate the primary power supply feeder breaker. However, the breaker number for the secondary power supply was given to the EO due to a misinterpretation of the annunciator response procedure. Upon investigation, the EO noted the breaker handle did not appear to be fully engaged in the closed position. The EO pushed the breaker handle towards the closed position to verify it was not tripped. At this point, the closed breaker contacts were momentarily interrupted resulting in a momentary power loss to the secondary power supply. As a result, the "A" main feedwater (MFW) regulating valve closed and both MFW pumps went to their low speed stop. The Reactor Operator attempted to take manual control of the MFW system, but was unsuccessful prior to the "A" steam generator reaching its low level reactor trip setpoint. All safety systems operated per design following the reactor trip. It was determined that procedures did not use standard terminology in referencing these power supplies. Personnel were also unaware of this molded case circuit breaker's inherent operational characteristic. The applicable procedure was clarified and other procedure revisions are under evaluation to standardize power supply terminology. Training will be provided to Operations personnel regarding this event.

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TEXT (If more space is required, use additional NRC Form 366A's)(17)

DESCRIPTION OF EVENT:

On November 26, 1999 at 1227 CST with the plant at 100% reactor power, a reactor trip occurred on low Steam Generator "A" level due to a momentary loss of power in control cabinet RP043. At 1204, control room annunciators alerted the control room staff of a power failure in control cabinet RP043. Initial investigations by I&C personnel revealed that the primary power supply (26 volt) in this cabinet was deenergized and the secondary power supply (24 volt) was supplying power to the cabinet. I&C personnel then requested that Operations verify the status of the feeder breaker to the primary power supply to determine if the feeder breaker had tripped. Control Room personnel referenced the annunciator response procedure to obtain the breaker number for the primary power supply. The annunciator procedure did not use the terminology of primary/secondary power supplies and instead referred to the supplies by their output voltage rating (i.e. +26 volt and +24 volt). Believing that the 24 volt power supply was the primary power supply, the Control Room staff directed the non-licensed Equipment Operator to verify the status of the feeder breaker to the 24 volt power supply.

Upon arrival at the 120 volt AC breaker, the Equipment Operator noted that the breaker handle did not appear to be fully engaged in the closed (on) position. To verify that the breaker was indeed closed, the Equipment Operator pushed the breaker handle towards the closed position and verified that it was not in the tripped position. At this point, the closed contacts of the 24 volt power supply feeder breaker were momentarily interrupted resulting in a momentary power loss to Control Panel RP043. As a result, the main feedwater regulating valve to the "A" steam generator closed and both main feedwater pumps went to their low speed stop. The licensed Reactor Operator attempted to take manual control of the "A" main feedwater regulating valve and both feedwater pumps, but was unsuccessful prior to the "A" steam generator reaching its low level reactor trip setpoint. As a result, a reactor trip occurred, a feedwater isolation signal was generated, and the auxiliary feedwater system actuated to supply the steam generators.

BASIS FOR REPORTABILITY:

This event is reportable per the requirements of 10 CFR 50.73(a)(2)(iv) to report automatic reactor protection system and engineered safety feature actuations.

CONDITION AT TIME OF EVENT:

Mode 1 – Power Operations - 100% Reactor Power

ROOT CAUSE:

It was determined that when the Equipment Operator pushed the breaker handle towards the closed position while the breaker was closed, the breaker contacts momentarily opened. This scenario was repeated on both the installed breaker following its replacement, and also on the replacement breaker prior to its installation.

(ITE Gould, model BQ, 120V, 30Amp.) Personnel were unaware of this inherent characteristic of this model breaker prior to this event.

As a result of these contacts opening, a momentary loss of power to the secondary power supply occurred, resulting in the de-energization of control cabinet RP043. As the control cabinet was re-energized, various controllers reset to manual control. This resulted in the closing of the "A" main feedwater regulating valve and the resetting of both main feedwater pump control circuits to their low speed stop setpoints.

Another causal factor associated with this event was attributed to the manner in which these power supplies are referenced within the annunciator response procedure. This procedure did not designate these power supplies as primary and secondary supplies to control cabinet RP043. Instead, these power supplies were identified by their output voltage rating, which led to the misunderstanding that resulted in the equipment operator being dispatched to the incorrect breaker.

The cause of the primary power supply failure was attributed to a defective fuse holder internal to the power supply. (Westinghouse 7300 Series power supply, part number 6005D54G01).

CORRECTIVE ACTIONS:

1) The primary power supply to this cabinet was replaced.

2) Procedure revisions are being evaluated to ensure that consistent/standard terminology is used throughout station procedures that reference these power supplies.

3) Training regarding this event and the operating characteristics of molded case circuit breakers will be provided to Operations personnel.

4) Industry notification of this molded case circuit breaker operational characteristic is being coordinated through the Institute of Nuclear Power Operations (INPO).

SAFETY SIGNIFICANCE:

The reactor automatically tripped per design due to low steam generator level. Plant safety features functioned as required and the radioactive material released to the environment as a result of this event was an insignificant fraction of the regulatory limits. This event was not significant with respect to the public health or safety.

PREVIOUS OCCURRENCES:

None.

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TEXT (If more space is required, use additional NRC Form 366A's)(17)

FOOTNOTES:

The system and component codes listed below are from IEEE Standard 805-1984 and IEEE Standard 803A-1983 respectively.

Systems	SJ, EC, JB, JK
Components	JX, BKR