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-4233

Ameren UE

Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSEE EVENT REPORT 1999-003-01
Manual Reactor Trip Due To Heater Drain System Pipe
Rupture Caused By Flow Accelerated Corrosion
(Reference: ULNRC-4106, dated September 10, 1999)

The enclosed licensee event report is submitted in accordance with 10CFR50.73(a)(2)(iv) to report an event that resulted in an automatic actuation of an Engineered Safety Feature and a manual 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

cc: Mr. Ellis W. Merschoff
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Jack N. Donohew (2 copies)
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop OWFN-4D3
Washington, DC 20555-2738

Manager, Electric Department
Missouri Public Service Commission
PO Box 360
Jefferson City, MO 65102

Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, GA 30339

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Callaway Plant Unit 1	0 5 0 0 0 4 8 3	1 OF 0 5

TITLE (4) **Manual Reactor Trip Due To Heater Drain System Pipe Rupture Caused by Flow Accelerated Corrosion**

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)
										0 5 0 0 0
0 8	1 1	9 9	9 9	- 0 0 3 -	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	MANUFAC-TURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO EPIX
E	S N	P S P		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)

At approximately 0917 on August 11, 1999 with the plant operating at 100 percent power, a pipe rupture occurred in a six inch drain line located between the "D" Moisture Separator Reheater first stage reheater drain tank and the "6B" feedwater heater. The reactor was manually tripped and fast closure of the Main Steam Isolation Valves was manually initiated. No personnel were injured during this event and all safety related systems operated per design. Some non-safety related equipment failures were observed for components located in the vicinity of the pipe rupture.

The location of the pipe rupture was determined to be immediately downstream of a 45 degree elbow. Root cause analysis indicates that the failure was attributed to Flow Accelerated Corrosion (FAC) with an abnormally high wear rate. The ruptured piping was replaced with Chromium - Molybdenum piping. Additional piping inspections were performed on locations with similar operating conditions/configurations with no other abnormal wear rates observed. Lessons learned from this event will be incorporated into FAC program guidelines for determining inspection points.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REV NO.	PAGE (3)
Callaway Plant Unit 1	05000483	99	- 003 -	01	02 OF 05

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

DESCRIPTION OF EVENT:

At approximately 0917 on August 11, 1999, a pipe rupture occurred in a six inch drain line located between the "D" Moisture Separator Reheater (MSR) first stage reheater drain tank and the "6B" feedwater heater. At the time of the event the plant was operating in Mode 1 at 100 percent power with no safety related components out of service.

Heater drain system alarms and an approximate 20 Megawatt decrease in electrical generation initially alerted the Control Room staff to an abnormality in the extraction steam/heater drain system. Subsequently, the Control Room began receiving reports of a steam leak within the Turbine Building. The Reactor Operator also noticed reactor power was increasing with a maximum observed power level of 100.78% (3591.8 MW thermal). The Control Room staff reduced reactor power and began making preparations to isolate one train of the high pressure feedwater heaters based on preliminary reports and indications. Additional reports indicated the steam leak was substantial and the source of the leak could not be determined. Based on this information, at 0925 the Control Room Supervisor directed the Reactor Operator to manually trip the reactor and the appropriate Emergency Operating Procedure was entered. The reactor trip initiated a main turbine trip which closed the turbine stop valves. Additionally, as a result of the reactor trip, a Feedwater Isolation Signal (FWIS) was generated which isolated the main feedwater system and automatically started the three Auxiliary Feedwater pumps to supply the steam generators.

In order to ensure isolation of the ruptured line, fast closure of the Main Steamline Isolation Valves was manually initiated with all valves closing properly. Personnel accountability by assembly was initiated. This was performed as a precautionary measure for ascertaining the safety of plant personnel.

The Control Room staff transitioned from the Emergency Operating Procedures to normal plant shutdown procedures and subsequently secured both motor driven auxiliary feedwater pumps. The plant was stabilized with the turbine driven auxiliary feedwater pump supplying the steam generators with the Main Steam Line atmospheric steam dump valves controlling steam generator pressure.

All safety related systems and components operated per design during this event. Operational abnormalities were observed for the following non-safety related systems and components following the rupture of the heater drain line:

1) Steam escaping from the ruptured pipe contacted fire detectors in the immediate area resulting in the actuation of a fire protection system preaction deluge valve. Although the deluge valve actuated, a discharge did not occur through the sprinkler heads since the fusible links on the sprinkler heads had not melted. However, the "A" diesel driven and motor driven fire pumps did automatically start as a result of the system pressure decrease from the deluge valve actuation.

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

2) The plant computer failed approximately 5 minutes into the event due to the failure of one of the computer's uninterruptible power supplies which was located in the vicinity of the pipe rupture.

3) The five non-safety process radiation monitors listed below experienced failures during this event due to the failure of an uninterruptible power supply (UPS) which was located in the vicinity of the pipe rupture.

Main Steam Line Atmospheric Steam Dump Valve "A" discharge monitor
Unit Vent discharge monitors "A" and "B"
Radwaste Building discharge monitors "A" and "B"

4) The in-service Condenser Air Removal Exhaust Fan tripped following the pipe rupture due to a high differential pressure on the suction filter resulting from the high humidity environment within the Turbine Building.

BASIS FOR REPORTABILITY:

These events are reportable per 10CFR50.73(a)(2)(iv) as an event or condition that resulted in a manual or automatic actuation of an engineered safety feature, including the reactor protection system.

CONDITION AT TIME OF EVENT:

Mode 1, Power Operations - 100% power

ROOT CAUSE:

The ruptured pipe was determined to be schedule 40, 0.280 inch wall thickness, SA106 Grade B carbon steel pipe. The normal operating conditions for this section of pipe was 410 degrees Fahrenheit, 283 psig. The location of the pipe rupture was determined to be immediately downstream of a 45 degree elbow in the heater drain line. The direct cause of the pipe rupture was preliminarily attributed to Flow Accelerated Corrosion (FAC) with an abnormally high wear rate. This was an unanticipated location for a catastrophic failure since the 45 degree elbow was considered to be the limiting FAC susceptible component in this location. Previous inspections had not revealed abnormal wear rates within this elbow.

Independent consultant services were acquired to assist in the root cause determination. Analysis has determined the FAC wear rate appears to have been accelerated by water droplet impingement on the piping immediately downstream of the 45 degree elbow. The cause of the impingement was attributed to a combination of the system's low steam quality (approximately 4%) and the piping configuration upstream of the 45 degree elbow. The upstream piping configuration consisted of a 100 foot horizontal section of pipe that allowed the water/steam mixture to separate, followed by 90 degree and 45 degree elbows. As the flow entered the 45 degree elbow, it encountered weld backing rings that forced some water droplets back into the steam flowpath. As the steam exited the 45 degree elbow, the water droplets were then impinged upon the side wall of the downstream piping.

A causal factor associated with this event is the manner in which the FAC inspection program provided guidance for inspecting downstream piping when performing fitting inspections. The program did not consider downstream piping inspections as mandatory, although it is a common practice. However, this downstream piping was not inspected when the upstream 45 degree elbow was inspected during previous refueling outages. Had this piping been previously inspected, it is probable an abnormal wear rate would have been detected. (Note: This section of pipe had been previously added to the scope of inspections for the plant's tenth refueling outage scheduled for the fall of 1999.)

A second causal factor associated with this event is the industry experience-based perception that the upstream 45 degree elbow would be the most susceptible component for failure due to FAC. This causal factor contributed to the decisions made during previous refueling outages to not inspect the downstream piping associated with this elbow.

CORRECTIVE ACTIONS:

1) Mechanical, electrical, instrumentation, and civil/structural inspections were performed to assess collateral damage to structures and components within the surrounding area of the pipe rupture.

2) The ruptured section of carbon steel piping was replaced with A335 Gr. P22 Chromium - Molybdenum piping.

3) Engineering performed evaluations to determine other locations with similar operating conditions/configurations which may be susceptible to the same failure mechanisms. As a result, 42 additional points within other sections of the heater drain system were identified as requiring inspections to verify piping integrity. These inspections were performed with no other abnormal wear conditions identified.

4) A review of components within the FAC program was performed in order to identify other locations in which downstream piping was not inspected during fitting inspections. Twenty-one inspection sites meeting these criteria were identified. As a result, these components have been added to the scope of inspections scheduled for the plant's tenth refueling outage in the fall of 1999.

5) Procedures governing the FAC program will be revised to include lessons learned from this event, including selection criteria guidelines for determining inspection points.

6) Information regarding this event has been provided for industry notification through the Checworks Users Group (CHUG), which is an industry funded users group that addresses FAC. In addition, utility engineering personnel have worked closely with the designers of the Checworks FAC modeling program in providing feedback regarding the specific details of this anomaly. General information regarding this event has also been provided to the Institute of Nuclear Power Operations (INPO) for industry distribution.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REV NO.	PAGE (3)
Callaway Plant Unit 1	05000483	99	- 003 -	01	05 OF 05

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

SAFETY SIGNIFICANCE:

No personnel injuries occurred as a result of this event. However, this event was significant with respect to the safety of plant personnel due to the potential for personnel injury resulting from the piping failure. This event was not significant with respect to the health and safety of the general public since all safety systems functioned per design during this event. Additionally, the radioactive material released to the environment as a result of this event was an insignificant fraction of the regulatory limits.

PREVIOUS OCCURRENCES:

There have been no previous occurrences of pipe ruptures due to FAC at Callaway.

FOOTNOTES:

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

System SN, Component PSP