AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

April 9, 2004

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

ULNRC04979

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSEE EVENT REPORT 2004-004-00
Safety Injection while conducting plant heatup to normal operating pressure and temperature.

Ladies and Gentlemen:

The enclosed licensee event report is submitted in accordance with 10CFR50.73(a)(2)(iv)(A) to report a Safety Injection while conducting plant heatup to normal operating pressure and temperature.

Sincerely,

Warren A. Witt

Warren A. Witt
Manager, Callaway Plant

Enclosure

IE22

Mr. Bruce S. Mallett
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

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 7E1
Washington, DC 20555-2738

Missouri Public Service Commission
Governor Office Building
200 Madison Street
PO Box 360
Jefferson City, MO 65102-0360

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

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Safety Injection while conducting plant heatup to normal operating pressure and temperature.

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
2	11	2004	2004 -	004 -	00	4	9	2004	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	3
10. POWER LEVEL	0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR : (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Mark A. Reidmeyer	(573) 676-4306

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	SB	TD	M120	Y					

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

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

At 2258, 2/11/04, with the plant in Mode 3, a Safety Injection (SI) occurred while performing a plant heat up to normal reactor coolant system operating pressure and temperature. The SI was the result of not performing a step contained in the procedure governing a plant heat up. All safety systems actuated as required and flow was initiated to the core due to plant conditions present at the start of the event. Emergency procedures were used to terminate the event and restore the plant to a normal condition. During the event, "B" Steam Generator Auxiliary Steam Dump (S/G ASD) did not properly operate and "A" Reactor Coolant Pump (RCP) exhibited high vibration. The "B" S/G ASD problem was identified as an obstruction of a balance arm within an electropneumatic transducer which was corrected and tested to verify operability. The RCP vibration was determined to be the result of thermal transients caused by the SI and did not require additional action.

A Root Cause Analysis investigation was conducted which revealed inadequate pre-job briefs, weaknesses in supervisory oversight, and cumbersome operating procedures as root causes for this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

This event is being reported per 10CFR50.73(a)(2)(iv)(A), system actuation.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

Callaway Plant was in Mode 3 performing a plant heat up to normal operating pressure and normal operating temperature (NOP/NOT).

C. STATUS OF STRUCTURES, SYSTEMS OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

None.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

At 1715, 2/11/04, a plant heat up was commenced using plant procedure OTG-ZZ-00001, PLANT HEATUP COLD SHUTDOWN TO HOT STANDBY. At 1748, Callaway transitioned from Mode 4 to Mode 3, Reactor Coolant System (RCS) Temperature Greater than 350 degrees F. Between 1730 and 1846, the On-Shift Control Room Operations staff conducted shift change turnovers. At the conclusion of the shift turnovers, a crew brief was conducted to ensure the control room staff was aware of existing plant status and plans for completion of plant heat up to NOP/NOT. Upon completion of the crew brief, the normal plant heat up continued using plant procedure OTG-ZZ-00001. As RCS pressure approached 1900 psig, OTG-ZZ-00001 contained steps to stabilize RCS pressure at 1900 psig using the Pressurizer (Pzr) pressure controller in automatic. Following this same sequence of steps was a CAUTION statement warning operators to ensure Steam Generator (S/G) pressure was greater than 615 psig prior to exceeding 1970 psig RCS pressure in order to prevent a Safety Injection (SI) on low steam line pressure.

At 2240, a licensed operator noticed that "A" Reactor Coolant Pump's (RCP) seal leak off trend was increasing. This unexpected indication distracted the operators as they discussed the problem. At approximately 2242, RCS pressure reached 1900 psig but no action was initiated by the control room operators to stabilize RCS pressure.

At 2258, RCS pressure reached 1970 psig, however, since RCS pressure had not been stabilized earlier at 1900 psig as procedurally required, steam generator pressure had only increased to approximately 600 psig, still below the SI setpoint of 615 psig. By design, with S/G pressure below the SI setpoint and RCS pressure reaching 1970 psig, a Safety Injection occurred. All equipment actuated as required and with the centrifugal charging pumps operating as High Head Injection, borated water was injected into the core.

Upon receipt of the SI, control room operators commenced a plant evaluation and recovery using approved plant emergency procedures. Emergency Action Level procedures were reviewed and it was determined that no emergency declaration was required. Operators began evaluation and recovery using emergency procedure E-0, REACTOR TRIP OR SAFETY INJECTION. As flow continued into the RCS, Pzr level and RCS pressure both began increasing. At 2306, Pzr Power Operated Relief Valves began cycling in response to increasing RCS pressure.

At 2312, control room operators transitioned to procedure ES 1.1, SAFETY INJECTION TERMINATION. At 2317, “A” and “B” SI pumps and “A” and “B” Residual Heat Removal pumps were secured, and plant recovery continued with the restoration of RCS letdown, securing of the “A” and “B” Emergency Diesel Generators, and use of “A”, “C”, and “D” S/G Atmospheric Steam Dumps (ASD) as the primary method of RCS heat removal. “B” S/G ASD was slow to respond to the demand signal and at 0002, 2/12/04, Technical Specification 3.7.4 was entered for an inoperable ASD.

At 0134, 2/12/04 the “A” RCP experienced high vibration and was secured. Further plant restoration continued without additional significant events. At 0251, the NRC Operations Center was contacted and Event Notification 40515 was filed to document this event.

E. METHOD OF DISCOVERY OF EACH COMPONENT, SYSTEM FAILURE, OR PROCEDURAL ERROR

Discovery of the Steam Line SI occurred when Main Control Board annunciators actuated.

A licensed operator identified the increasing “A” Reactor Coolant Pump’s (RCP) vibration trend.

Improper response of “B” S/G ASD was observed by a licensed operator monitoring indications on the Main Control Boards.

II. <u>EVENT DRIVEN INFORMATION</u>

A. SAFETY SYSTEMS THAT RESPONDED

The following systems actuated as a result of the SI:

- High Head Injection (Centrifugal Charging pumps)
- Intermediate Head Injection (Safety Injection pumps)
- Low Head Injection (Residual Heat Removal pumps)
- Auxiliary Feedwater
- Essential Service Water
- Emergency Diesel Generators

The following isolation signals were also generated:

- Control Room Ventilation Isolation Signal (CRVIS)
- Containment Purge Isolation Signal (CPIS)
- Containment Isolation Phase “A” (CISA)

B. DURATION OF SAFETY SYSTEM INOPERABILITY

Not applicable to this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT.

A probabilistic risk assessment (PRA) determined that the reported event was of very low risk significance.

III. <u>CAUSE OF THE EVENT</u>

<u>Safety Injection:</u>

A multi-disciplinary Root Cause Analysis (RCA) team was assembled to investigate this event. This RCA

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team was conducted for both LER 2004-004-00 and 2004-005-00. These two events were found to have related root causes and corrective actions. The RCA team conclusions for this event were:

Root Cause 01 (RC-01)
Policy Guidance regarding Pre-Job Briefs is not strict enough and allows interpretation, resulting in varying degrees of quality of Pre-Job briefs.

Causal Factors identified by the Root Cause Analysis (RCA) (RC-01)

Operations had a mindset that heat up and cool down evolutions were simple and controlled, and a higher degree of situational awareness was not necessary (e.g. heatup to NOP/NOT). Because of this mindset, a thorough task preview, pre-job briefs (PJB) and Pre-evolution Practice (PREP) were not performed. Risk factors of an infrequently performed evolution or the experience level of the crew did not prompt the need for a thorough task preview, pre-job brief and PREP.

Root Cause 02 (RC-02)
Operations supervisory oversight and standards reinforcement needs improvement.

Causal Factors identified by the Root Cause Analysis (RCA) Team (RC-02)

The Shift Supervisor (SS) and Control Room Supervisor (CRS) need to provide more oversight when there is an infrequent evolution. Instead of providing oversight, the CRS was distracted with multi-tasking and look-ahead activities.

Root Cause 03 (RC-03)
General Operating procedures (OTGs) are cumbersome and difficult to follow.

Causal Factors identified by the RCA (RC-03)

The crew did not perform OTG-ZZ-00001 steps in 6.4.13. The steps contained a critical step which infers to "STOP" at 1900 psig RCS pressure and put the pressurizer pressure controller in automatic. The OTGs are sequenced poorly and have a confusing layout, making placekeeping difficult. The procedure also contains ambiguous terms such as "if applicable" and "as needed" without criteria to their application.

RCP high vibration:

Investigation concluded that the vibration noted was thermally induced due to changing conditions within the RCS. Vibration levels had not demonstrated a marked trend over time and this occurrence did not warrant additional actions.

"B" S/G ASD:

Investigation determined that an electrical wire had become positioned behind a balance arm located inside the electropneumatic transducer (manufactured by Masoneilan; Model 8005N). This physical impediment of the balance arm resulted in a slow response to a demand signal. By repositioning the wire, proper operation was restored and the ASD declared Operable.

IV. CORRECTIVE ACTIONS

The following Corrective Actions were developed by the RCA team. Corrective Actions to Prevent Reoccurrence (CATPR) are actions that will be taken in order to prevent a similar event from occurring in

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the future. Corrective Actions (CA) are additional actions recommended by the RCA team that should be evaluated for incorporation but are not required to be implemented. Not all Root Causes will have CA.

CATPR 01
Expectations for Pre-Job Briefs (PJB) have been strengthened in Operations. The Plant Manager and Superintendent of Operations conducted briefings with each crew affirming the expectations for pre-job briefs. The Shift Supervisors and Operations management conducted an all-day performance review meeting where the expectation for performing pre-job briefs was reaffirmed. In addition, the Senior Reactor Operators met as a group and discussed the importance and expectations of pre-job briefs.

Improved site-wide guidance for pre-job briefs is being evaluated for implementation.

CATPR 02
The Shift Supervisor and Senior Reactor Operator meetings discussed under CATPR 01 also emphasized the following Operations standards:
- Ensure the level of supervisory oversight compensates for infrequency of evolutions, experience level of crew members, and evolutions of high consequence.
- Ensure standards address the roles and responsibilities of crew members and supervisors.
- Ensure leadership is engaged and is conducting observations and coaching personnel.

CATPR 03
Revise OTG-ZZ-00001 to address layout and formatting concerns, and sequencing of steps.

CA-01
Evaluate adding annunciators to alert Control Room Operators when conditions are imminent for Low Steam Line Pressure Safety Injection.

V. PREVIOUS SIMILAR EVENTS

A review of past LERs between 2/3/01 and 2/3/04 confirmed that there were no additional LERs submitted due to a Safety Injection event.
A review of the Callaway Action Request System did not disclose any CARs reporting Safety Injection actuations due the time period between 2/3/01 and 2/3/04.

VI. ADDITIONAL INFORMATION

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

Safety Injection:
System: BQ

Component: Not applicable – this event was not resultant from an equipment failure.

"B" S/G ASD:
System: SB

Component: TD