UNION ELECTRIC

Callaway Plant

February 24, 1992

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

ULNRC-2568

Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
FACILITY OPERATING LICENSE NPF-30
LICENSEE EVENT REPORT 92-003-00
ENGINEERED SAFETY FEATURE (ESF) ACTUATION
ON HIGH STEAM GENERATOR (S/G) LEVEL
DUE TO FEEDWATER CONTROL PROBLEMS
WHILE MAINTAINING A LOW POWER LEVEL

The enclosed Licensee Event Report is submitted pursuant to 10CFR50.73(a)(2)(iv) concerning an Engineered Safety Feature Actuation and a Turbine Trip on high steam generator level during secondary plant startup.

J. D. Blosser
Manager, Callaway Plant

JDB/TPS/MNF/lrj

Enclosure

cc: Distribution attached

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NRC FORM 366 (6-89) *

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

TITLE (4): Engineered Safety Feature (ESF) Actuation On High Steam Generator (S/G) Level Due To Feedwater Control Problems While Maintaining A Low Power Level

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 1	2 3	9 2	9 2	0 0 3	0 0	0 2	2 4	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	1 1 5	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Thomas P. Sharkey, Supervising Engineer, Site Licensing	AREA CODE 3 1 4 6 7 6 - 8 3 3 6

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	

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 1/23/92 at 2259 CST, a feedwater isolation and auxiliary feedwater actuation Engineered Safety Features (ESF) actuations were received due to a high level signal in steam generator (S/G) 'D'. The plant was in Mode 1 at 15%.

At 2220 CST, a plant startup was in progress with the main turbine synchronized. Reactor Operators (RO) were attempting to maintain turbine load and reactor power at 15% to avoid accumulating Axial Flux Difference penalty deviation time per Technical Specification (T/S) 3.2.1 while performing surveillance test, OSP-AC-00004, Main Turbine Overspeed Test.

The RO manually closed the four feedwater bypass control valves and attempted to control the S/G level oscillations using the main feedwater regulator valves. Procedure OSP-AC-00004 was completed and the RO began increasing main generator load. Oscillations in the S/G levels increased and S/G 'D' exceeded its high level setpoint, causing a turbine trip and the ESF actuations. The plant was returned to Mode 1 at 1002 on 1/24/92 and power ascension was resumed.

The root cause was the attempt to maintain reactor and turbine power at low loads (</=15% power). To prevent recurrence, procedure OSP-AC-00004 will be performed between 20% and 49% power. The approval of a submitted Operating License amendment request to revise T/S 3.2.1 will eliminate the concern of accumulating flux penalty deviation time at a low power.

BASIS FOR REPORTABILITY:

On 1/23/92 at 2259 CST, an Engineered Safety Feature (ESF)(1) Feedwater Isolation (FWI) and an Auxiliary Feedwater Actuation (AFA) occurred due to the water level in the 'D' Steam Generator(2) (S/G) increasing to the high level trip setpoint. This report is submitted pursuant to 10CFR50.73(a)(2)(iv) to report an event which resulted in the automatic actuation of Engineered Safety Features.

CONDITIONS AT THE TIME OF EVENT:

Mode 1 - Power Operation
Reactor Power - 15%
Reactor Coolant System temperature -564 degrees F
Reactor Coolant System pressure - 2235 psig

DESCRIPTION OF EVENT:

On 1/23/92, a plant startup was in progress. At 2220 CST, the main turbine(3) was synchronized to the electrical system and loaded to approximately 120 MegaWatts electrical. A utility Senior Reactor Operator (SRO) trainee under the direct supervision of a licensed Reactor Operator (RO) was controlling the S/G levels with the 'A' main feedwater pump(4) in manual control and the feedwater bypass control valves(5) in automatic control. At approximately 2225, procedure OSP-AC-00004, Main Turbine Overspeed Test, was begun per the surveillance Mode Change Letter (MCL). The completion of OSP-AC-00004 was delayed while replacing turbine control panel(6) light bulbs that were necessary to verify procedure steps. S/G levels began to oscillate during this time delay. Between 2244 and 2254, the Balance of Plant (BOP) RO manually closed the four feedwater bypass control valves and attempted to control the S/G level oscillations using the main feedwater regulator valves. During this time, the operators were also attempting to maintain turbine load and reactor power at 15% to avoid accumulating Axial Flux Difference penalty deviation time.

Procedure OSP-AC-00004 was completed at 2255 and the operators began increasing the main generator load. S/G level oscillations increased and S/G 'D' exceeded its high level trip setpoint on 1/23/92 at 2259 CST. A FWI, AFA and turbine trip signals were generated by design. The plant stabilized in Mode 2 at 2% reactor power. The operators immediately verified proper operation of the ESF feedwater isolation valve closures(7) and the motor driven Auxiliary Feedwater pump(8) starts. They restarted the 'A' main feedwater pump, secured auxiliary feedwater, and reset the main turbine generator in accordance with plant procedures. The plant was returned to Mode 1 at 1002 on 1/24/92 and power ascension was resumed.

ROOT CAUSE:

The root cause was the attempt to maintain reactor and turbine power at low loads (</=15% power) for an extended period of time. The power level was being maintained less than or equal to 15% since the operators were concerned about limiting the anticipated time out of the Axial Flux Difference Band. This results in penalty deviation time per Technical Specification 3.2.1 and a reactor power restriction to less than 50% if more than one hour of penalty time is accumulated. S/G water level oscillations began during this time and the operators were not successful in reducing the S/G oscillations prior to the S/G high level ESF actuation. The following items contributed to the time delay at low power: the performance of the OSP-AC-00004 surveillance test was imposed on the operators by the licensed shift supervisor at the last minute prior to exceeding 15% power in order to comply with an overly restrictive MCL (the surveillance could have been performed at any power level prior to 50% power); and completion of OSP-AC-00004 was delayed (by approximately 15 minutes) due to the need to replace burned out light bulbs on the turbine control panel. Due to the extended time at low power, the feedwater temperature decreased from approximately 400 degrees F to 290 degrees F. This temperature drop was caused by the need for increased feedwater flows without a comparable increase in feedwater preheating at low power. The addition of cold feedwater to the hot S/G's further perturbated the existing oscillations of the S/G water levels.

As a contributing factor, the licensed operators were supervising three SRO trainees during the startup. The primary RO supervised one trainee. The two licensed BOP RO's supervised a trainee controlling S/G levels and one attending to the main generator. The control room licensed operators did not direct the trainees to step away from the controls when the S/G level oscillations began.

CORRECTIVE ACTIONS:

1. The following procedure and documents will be reviewed and revised as necessary to ensure surveillance OSP-AC-00004, Main Turbine Overspeed Test, will be consistently performed between 20% and 49% reactor power: a) Procedure OTG-ZZ-00004, General Operating Procedure, Power Operation, b) the Surveillance Task Sheet for performing surveillance OSP-AC-00004; and c) Mode Change Letter.

2. Bulb checks will be performed on the turbine control panel prior to synchronizing the main turbine generator to the grid in order to avoid delays in overspeed testing.

3. An Operating License amendment request (ULNRC-2546 dated 1/14/92) was submitted to revise Technical Specification 3.2.1, Axial Flux Difference. This proposal changes the mode applicability to Mode 1 above 50% of Rated Thermal Power following a change to Relaxed Axial Offset Control operation. When approved, the concern of accumulating flux penalty deviation time at low power will be eliminated.

4. The following policies have been implemented: a) The number of trainees allowed in the control room has been limited; b) Specific guidance has been provided for trainee actions during a plant transient.

SAFETY SIGNIFICANCE:

The ESF systems performed as designed in response to the S/G water level. There were no detrimental effects on any plant equipment as a result of the FWI and AFA actuations. This event had no adverse affect on the public health and safety.

PREVIOUS OCCURRENCES:

The following events were similar to the current event in that the ESF actuations occurred due to a S/G level exceeding the high level trip setpoint. However, these previous occurrences involved some mechanical failures which created plant conditions different than this reported event. Additionally, plant startup procedures were significantly revised after these previous events and have greatly improved S/G water level controls at a low power level.

LER 85-012-00; ULNRC-1066 dated 3/25/85.
LER 88-005-00; ULNRC-1773 dated 5/16/88.
LER 88-010-00; ULNRC-1838 dated 10/3/88.

FOOTNOTES:

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

1. System - JE
2. System - AB, Component - SG
3. System - TA
4. System - SJ, Component - P
5. System - SJ, Component - FCV
6. System - IT, Component - IL
7. System - SJ, Component - ISV
8. System - BA, Component - P