# ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8811010085 DOC.DATE: 88/10/20 NOTARIZED: NO DOCKET #
FACIL:50-335 St. Lucie Plant, Unit 1, Florida Power & Light Co. 05000335

| AUTH.NAME | AUTHOR AFFILIATION |
|---|---|
| SNYDER,M.J. | Florida Power & Light Co. |
| CONWAY,W.F. | Florida Power & Light Co. |
| RECIP.NAME | RECIPIENT AFFILIATION |

SUBJECT: LER 88-008-00:on 880920,reactor trip on low steam generator level due to inadvertent closure of main feedwater.
W/8 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 5
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

| | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL | RECIPIENT ID CODE/NAME | COPIES LTTR | COPIES ENCL |
|---|---|---|---|---|---|---|
| | PD2-2 LA | 1 | 1 | PD2-2 PD | 1 | 1 |
| | TOURIGNY,E | 1 | 1 | | | |
| INTERNAL: | ACRS MICHELSON | 1 | 1 | ACRS MOELLER | 2 | 2 |
| | ACRS WYLIE | 1 | 1 | AEOD/DOA | 1 | 1 |
| | AEOD/DSP/TPAB | 1 | 1 | ARM/DCTS/DAB | 1 | 1 |
| | DEDRO | 1 | 1 | NRR/DEST/ADS 7E | 1 | 0 |
| | NRR/DEST/CEB 8H | 1 | 1 | NRR/DEST/ESB 8D | 1 | 1 |
| | NRR/DEST/ICSB 7 | 1 | 1 | NRR/DEST/MEB 9H | 1 | 1 |
| | NRR/DEST/MTB 9H | 1 | 1 | NRR/DEST/PSB 8D | 1 | 1 |
| | NRR/DEST/RSB 8E | 1 | 1 | NRR/DEST/SGB 8D | 1 | 1 |
| | NRR/DLPQ/HFB 10 | 1 | 1 | NRR/DLPQ/QAB 10 | 1 | 1 |
| | NRR/DOEA/EAB 11 | 1 | 1 | NRR/DREP/RAB 10 | 1 | 1 |
| | NRR/DREP/RPB 10 | 2 | 2 | NRR/DRIS/SIB 9A | 1 | 1 |
| | NUDOCS-ABSTRACT | 1 | 1 | REG FILE 02 | 1 | 1 |
| | RES/DSIR/EIB | 1 | 1 | RES/DSR DEPY | 1 | 1 |
| | RES/DSR/PRAB | 1 | 1 | RGN2 FILE 01 | 1 | 1 |
| EXTERNAL: | EG&G WILLIAMS,S | 4 | 4 | FORD BLDG HOY,A | 1 | 1 |
| | H ST LOBBY WARD | 1 | 1 | LPDR | 1 | 1 |
| | NRC PDR | 1 | 1 | NSIC HARRIS,J | 1 | 1 |
| | NSIC MAYS,G | 1 | 1 | | | |

TOTAL NUMBER OF COPIES REQUIRED: LTTR 43 ENCL 42

R I D S / A D D S

R I D S / A D D S

A104

NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| St. Lucie Unit 1 | 0 5 0 0 0 3 3 5 | 1 OF 0 4 |

TITLE (4) REACTOR TRIP ON LOW STEAM GENERATOR LEVEL DUE TO INADVERTENT CLOSURE OF A MAIN FEEDWATER REGULATING VALVE

| 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 9 | 2 0 | 8 8 | 8 8 | — 0 0 8 | — 0 0 | 1 0 | 2 0 | 8 8 | N/A | 0 5 0 0 0 |
| | | | | | | | | | N/A | 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)

| | | | |
|---|---|---|---|
| [ ] 20.402(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 |
|---|---|
| M. J. Snyder, Shift Technical Advisor | AREA CODE 4 0 7 4 6 5 - 3 5 5 0 |

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

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS |
|---|---|---|---|---|---|---|---|---|---|
| X | I U | M O N | E 1 4 6 | Yes | | | | | |
| | | | | | | | | | |

| 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)

Prior to the event on 20 September, 1988, Unit 1 was at 100% power, steady state operations. Instrumentation & Control (I&C) personnel were working on the 'B' Steam Generator Feed Regulating System (SG FRS) in an effort to minimize water level swings of 4% narrow range indication in the 'B' Steam Generator (SG). At 1152, a power supply wire lead, which was not shown on the vendor wiring diagram being used, was inadvertently lifted when I&C correctly removed another lead on the same terminal lug connection. Removal of the power supply lead deenergized the 'B' FRS control circuit, and caused the valve to shut. The loss of half of normal main feedwater supply caused SG water level to rapidly decrease. The unit was tripped on low SG level by the Reactor Protective System and the Reactor Operator. Corrective actions were to upgrade the administrative procedure used, install a double lug terminal connection in the FRS circuitry to preclude accidental lead lifting, and to evaluate and repair the root cause of SG level swings before resuming power operations.



IE22
1/1

DESCRIPTION OF EVENT:

Prior to the reactor trip on 20 September, 1988, Unit 1 was at 100% power, steady state operations. At the time of the event, all systems required for safe shutdown were operable. All automatic controllers were in automatic except for the Control Rod Drive System (EIIS:JD) which was in "OFF" and 'B' Steam Generator (SG) level control (EIIS:JB) which was in manual.

Since startup from the last refueling outage, August 31, 1988, 'B' Steam Generator's (SG) water level exhibited peak to peak swings of four percent on the narrow range indication. Plant personnel suspected that the cause of these SG level swings was a faulty lead/lag circuit in the 'B' SG FRS controller. Instrumentation & Control (I&C) personnel proposed to minimize these level swings by lifting leads in the lead/lag circuitry in the FRS controller. The proper administrative controls for lifting the lead were followed.

The evolution called for the removal of one wire lead from a terminal connection. It was noted that an additional wire lead was secured to this particular terminal connection, and it was not depicted on the vendor's wiring diagram of the circuitry. Work proceeded and the correct wire lead was lifted from the terminal connection at 1152. The second wire lead, now unsecured to the terminal connection, immediately came off the terminal connection as well. This lead was the power supply to controller of the 'B' FRS. Loss of power to the FRS caused the 1B Main Feed Regulating Valve (MFRV) (EIIS:SJ) to shut.

Licensed operators who were controlling 'B' SG level were immediately alerted to the closure of the 'B' MFRV and decreasing SG level, and called to I&C personnel to restore what they had just done. In less than 30 seconds, licensed operators decided to trip the unit when it became apparent that it was impossible to recover from the SG water level transient. A manual reactor trip was preceded by an automatic reactor trip on low SG level (EIIS:JC) by less than one second.

The trip was uncomplicated. The newly installed Control Element Assembly Position Display system (EIIS:IU), erroneously indicated that several control rods were not in their fully inserted position. The core mimic was used to verify that all rods were on the bottom after the unit trip. The Steam Bypass Control System (EIIS:JI) operated to reduce primary average temperature to the zero power setpoint of 532 degrees F. The Auxiliary Feedwater Actuation System (AFAS) (EIIS:BA) actuated as expected on low SG level due to the partial loss of main feedwater. The standard post trip actions were completed by the operators and the unit was quickly stabilized in a hot standby condition.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| St. Lucie Unit 1 | 0 5 0 0 0 3 3 5 | 8 8 | — 0 0 8 — | 0 0 | 0 3 OF 0 4 |

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

CAUSE OF THE EVENT:

The root cause for this event was that the procedure governing the activity did not caution the maintenance personnel to stop and evaluate conditions when the actual wiring was not as expected. (The wiring diagram used in this event and the field conditions were electrically the same, but that the exact terminal and wiring conditions differed slightly.) A step by step following of the existing maintenance procedure did not address unexpected field conditions. As a result, the literal following of the procedure resulted in the reactor trip. There were also several contributing factors to this event. Maintenance personnel did not stop work to consider what actions should be taken when wiring diagram and field differences are found. Also, the design placement of two wiring leads onto one terminal board connection made it difficult to maintain a good electrical contact on the bottom lead while removing the top lead. In summary, the root cause of this event was due to a procedural deficiency. Contributing factors were: equipment design and a personnel error in judgement.

There were no adverse environmental working conditions which contributed to this event.

ANALYSIS OF EVENT:

This event is considered reportable under 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System."

The inadvertent removal of power from the 'B' FRS controller caused the MFRV to shut, the resultant partial loss of feedwater caused a reactor trip on low SG level within seconds of the event's initiation. Both the RPS and a licensed operator tripped the unit within one second of each other on a low SG level. An AFAS signal was generated on low SG level, and the Auxiliary Feedwater system functioned as designed to automatically supply condensate to both SGs. The Steam Bypass Control System functioned as designed to control primary temperature in tandem with licensed utility operator actions to regulate SG water levels and primary cooldown rate with auxiliary feedwater.

A system which did not function as expected during the event was the newly installed Control Element Assembly Position Display System (CEPDS). After the trip, this system graphically depicted at least one control rod in each group to be not fully inserted. To verify the complete insertion of all control rods after the trip, licensed operators used the safety related core mimic display which showed all rods to be inserted. Using the core mimic, which like CEPDS uses reed switch position indication, is one of two methods for verifying full rod insertion. The other method for control rod position verification which does not use reed switches is by using the Digital Data Processing System. Additionally, reactivity control is verified during each trip by operator observation of decreasing reactor power. I&C personnel determined that the erroneous indication given by CEPDS was due to an electrical ground in the system. The fault was corrected before the unit was restarted.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | PAGE (3) |
|---|---|---|---|---|---|
| St. Lucie Unit 1 | 0 5 0 0 0 3 3 5 | 8 8 | — 0 0 8 — | 0 0 | 0 4 OF 0 4 |

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

ANALYSIS OF EVENT: (Continued)

Chapter 15.2.8 of the St. Lucie Unit 1 FSAR analyzes a loss of normal feedwater flow using more limiting initial conditions than that experienced during this event. A comparison of this event with that analysis shows that no violation of reactor safety limits occurred. Therefore, at no time during the event were the health and safety of the public endangered.

CORRECTIVE ACTIONS:

1. Following the trip the unit was stabilized in hot standby.
2. Before the unit was returned to power, extensive testing of the 'B' MFRV revealed the root cause of the SG level swings to be a sticking actuator on the valve. The actuator was replaced and SG levels are steady.
3. Identified connections in the FRS which have two wire lead terminations will be double lugged to preclude inadvertently lifting the bottom lead. This will be done for both units during the next scheduled unit shutdown.
4. A procedure change will caution maintenance personnel in working on field circuitry that differs from what is expected.
5. Utility personnel were counseled on the need to stop work and analyze consequences when field conditions are not consistent with expected conditions.
6. A Human Factors Evaluation Report has been generated.

ADDITIONAL INFORMATION:

Previous Similar Events

See LER #335-88-003 for a similar previous reactor trip event.

Component Failures

There were no material or component failures which adversely contributed to this event. The reed switches that failed and caused the erroneous CEA position indications are model N9020 manufactured by Electro-Mechanics Inc.

OCTOBER 2 0 1988

L-88-457
10 CFR 50.73

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: St. Lucie Unit 1
Docket No. 50-335
Reportable Event: 88-08
Date of Event: September 20, 1988
Reactor Trip on Low Steam Generator Level Due to
Inadvertent Closure of a Main Feedwater Regulating Valve

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

W F Conway

W. F. Conway
Senior Vice President - Nuclear

WFC/GRM/cm

Attachment

cc: Malcolm L. Ernst, Acting Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

GRMLER08

IE22
1/1