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Ref: 10CFR50.73(a)(2)(iv)

W3B5-91-0210
A4.05
QA

August 5, 1991

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Submittal of Licensee Event Report

Gentlemen:

Attached is Licensee Event Report Number LER-89-024-01 for Waterford Steam Electric Station Unit 3. This Licensee Event Report supplement is submitted to provide additional information on corrective action resulting from the investigation of the event described. This Licensee Event Report is submitted pursuant to 10CFR50.73 (a)(2)(iv).

Very truly yours,

D.F. Packer
General Manager - Plant Operations

DFP/LDC/jrr
Attachment

cc: Messrs. R.D. Martin
G.L. Florreich
J.T. Wheelock - INPO Records Center
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NRC Resident Inspectors Office



IE22
1/1

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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.

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Waterford Steam Electric Station Unit 3 | 0 5 0 0 0 3 8 2 | 1 OF 0 6 |

TITLE (4)
Reactor Trip due to Loss of Feedwater Flow to Steam Generator #1

| 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) |
| | | | | | | | | | | | N/A | 0 5 0 0 0 |
| 1 2 | 2 3 | 8 9 | 8 9 | - | 0 2 4 | - | 0 1 | 0 8 | 0 5 | 9 1 | N/A | 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 0 0 | 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 |
|---|---|
| R.S. Starkey, Operations and Maintenance Manager | AREA CODE: 5 0 4 — 4 6 4 - 3 1 3 4 |

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)


At 1109 hours on December 23, 1989, control room operators initiated a manual reactor trip of Waterford Steam Electric Station Unit 3 while operating at 100% power. The trip was initiated in response to decreasing level in Steam Generator (SG) #1 after Main Feed Regulating Valve (MFRV) #1 unexpectedly failed shut. Shortly after the reactor trip, MFRV #1 opened inadvertently. A Reactor Coolant System (RCS) cooldown and a corresponding RCS pressure drop to approximately 1640 psia resulted, generating a Safety Injection Actuation Signal (SIAS). An Emergency Feedwater Actuation Signal (EFAS) was also generated during the post-trip transient.

The root cause of this event was an anomaly in the MFRV pneumatic control system aggravated by cold weather effects on system components. A degraded diaphragm in the MFRV volume booster led to the inadvertent shutting of MFRV #1. All safety systems functioned as designed; therefore, this event did not threaten the health and safety of the general public or plant personnel.

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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-630), 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) | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| Waterford Steam Electric Station Unit 3 | 0 5 0 0 0 3 8 2 | 8 9 | — 0 2 4 | — 0 1 | 0 2 OF 0 6 |

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

At 1103 hours on December 23, 1989, Waterford Steam Electric Station Unit 3 was operating at 100% power when a Steam Flow/Feedwater Flow signal deviation alarm (EIIS Identifier IB-FFA) was received for both steam generator's (SG) (EIIS Identifier-SG). SG #1 level was observed decreasing with SG #2 level observed increasing. Control room operators assumed manual control of both main feedwater regulating valve (MFRV) (EIIS Identifier SJ-FCV) and SG Feed Pump (SGFP) (EIIS Identifier SG-P) controllers (EIIS Identifier-FCO). MFRV #2 responded normally to operator input signals; however, MFRV #1 responded sluggishly by cycling between 5.0E6 and 8.0E6 lbm/hr. At 1109 hours, MFRV #1 unexpectedly failed shut and would not respond to manual input signals. Control room personnel tripped the reactor as SG #1 level approached its reactor protection system (RPS) (EIIS Identifier-JC) low level trip setpoint, preempting a challenge to the RPS.

During the minute following the reactor trip, MFRV #1 opened to approximately 40%, inducing a reactor coolant system (RCS) (EIIS Identifier-AB) cooldown and corresponding RCS pressure decrease. RCS pressure decreased below the safety injection actuation signal (SIAS) (EIIS Identifier-JE) setpoint of 1684 psia (lowest pressure reached 1640 psia). All safety injection (SI) system (EIIS Identifier-BP/BQ) components started as designed; however, no SI flow was injected into the RCS. Also initiated during the minute following the reactor trip was an emergency feedwater actuation signal (EFAS) (EIIS Identifier-JE), which started emergency feedwater system (EFW) (EIIS Identifier-BA) components. After taking manual control of MFRV #1, control room personnel were able to shut MFRV #1, gain control of SG level and RCS pressure and stabilize plant conditions in Mode 3 (Hot Standby).

Earlier on December 23, 1989, at 0448 hours, a Steam Flow/Feedwater Flow signal deviation alarm was received and SG #2 level was observed to be increasing. Operators took manual control of MFRV #2 and SGFP controllers and were able to stabilize SG level. Instrumentation and Control (I&C) technicians were called in to investigate the problem but did not identify any abnormalities in the associated valve control circuitry. MFRV and SGFP control was placed in automatic at 0815 hours.

The root cause of the event was an anomaly in the MFRV pneumatic positioning system aggravated by cold weather. The MFRV's are 16-inch angle globe valves controlled by an electro-pneumatic control system. An electrical signal from the feedwater control system (FWCS) (EIIS Identifier-JB) is converted to a pneumatic signal in the electro-pneumatic (E/P) converter (EIIS Identifier-CNV) which in turn supplies instrument air (IA) (EIIS Identifier-LD) to the valve positioner (Fisher Controls Inc., Model 3570). Also in the pneumatic control system is an air regulator (EIIS Identifier-RG), volume booster relays, solenoid valves (EIIS Identifier SOL-V), pneumatic valves, and a check valve. The majority of these components employ the use of diaphragms (primarily Buna-N and Viton) of which the most limiting are designed for temperatures in the range of 0 to 150 degrees F. At the time of the MFRV failures, ambient temperatures were 12-15 degrees F. These components are externally mounted and were subjected to abnormally low temperatures. Diagnostic testing and examination of a volume booster diaphragm indicated that degraded performance of the volume booster diaphragm caused the failure of MFRV #1.

Another cause investigated was potential ice particle formation in the IA system which could have clogged small air flow passages in the positioner or some other component. Valve filter regulator blowdowns conducted after the trip did not indicate any moisture. Also, shiftly blowdowns of system low points had not indicated moisture content prior to or after this event. Dewpoint indications have routinely been well within allowed levels (-20 degrees F dewpoint alarm setpoint). The IA task force reexamined practices to ensure that adequate measures exist to preclude the presence of moisture in the IA system and concluded that moisture in the IA system was not a factor in this event. A chemical analysis performed on the volume booster diaphragm supports this conclusion and also indicated that no chemical contaminants were present in the system.

It is important to note that within 30 minutes of both MFRV abnormalities on December 23 (0448 and 1103 hours), similar problems were observed with #1 High Pressure (HP) Feedwater Heater level control valve (LCV) (EIIS Identifier SJ-LCV) and Moisture Separator Reheater (MSR) temperature control valves (TCVs) (EIIS Identifier SB-TCV). HP Feedwater Heater LCVs and MSR TCVs utilize a similar pneumatic operating system, further indicating that the low ambient temperatures experienced were a possible contributor to the faulty valve operation.

The RCS cooldown and subsequent SIAS was caused by MFRV #1 opening independent of operator action. While attempting to gain control of MFRV #1 after it had inadvertently shut, the operator initiated a demand signal to open MFRV #1. Unresponsive to this demand signal, MFRV #1 remained shut and the reactor was tripped. After tripping the reactor, the operators properly carried out actions required by plant procedures which included verifying the MFRVs shut. The operator assumed MFRV #1 had failed closed. No consideration was given for an intermittent failure that would immediately correct itself. When the effects of the anomaly that caused MFRV # 1 to shut subsided, MFRV #1 operated according to its demand signal (approximately 40% open). This feedwater flow together with a failure of MSR #2 TCVs to fully shut caused the RCS cooldown that followed the trip. Although not specifically required by Emergency Operating Procedures, entering a 0% demand signal to the FWCS manual/automatic (M/A) station as a precautionary measure would have prevented MFRV #1 from opening. Because of the nature of this problem, the consideration of a potential intermittent FWCS failure was added to operator initial and requalification training programs to prevent future occurrences.

Immediately following the trip a tent was erected around the MFRVs and portable heaters were installed to remove the effects of the low temperatures. After reactor startup with local temperatures still in the teens no further problems were observed with MFRV control.

Plant systems (including SI and EFW) were aligned for startup and the plant was placed back on the grid at 0634 hours on December 24, 1989. To summarize, the following actions have been initiated or completed to identify the cause and prevent recurrence:

1. The plant freeze protection procedure was evaluated for possible enhancements that could preclude future cold weather induced equipment malfunctions. Several enhancements have been incorporated into plant freeze protection procedures.

2. The IA task force was reassembled to assess current practices pertaining to maintaining a moisture free IA system and concluded that no major changes in IA system practices were required.

3. Vendor assistance was obtained during a short plant outage to perform diagnostic testing of the MFRVs to determine the root cause.

4. Instructions were incorporated into operator initial and requalification training to verify FWCS in automatic or to input a zero demand signal into the FWCS M/A station following a trip as a precautionary measure when in manual control.

5. Diagnostic testing on the MFRVs will be implemented on an 18 month interval. This testing will identify if the valves require soft goods replacement.

Because all safety systems functioned as designed, there was no threat to the health and safety of the general public or plant personnel during this event.

SIMILAR EVENTS:

LER 85-029 described a reactor trip which was caused by high SG level due to MFRV positioners being out of adjustment but was not cold weather related.

PLANT CONTACT

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