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FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269

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SUBJECT: LER 89-001-00:on 890102,reactor trip due to personnel error.
W/8 ltr.

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EXPIRES: 8/31/85

# LICENSEE EVENT REPORT (LER)

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TITLE (4)
Reactor Trip Due to Personnel Error

| 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 | 0 2 | 8 9 | 8 9 | — 0 0 1 — | 0 0 | 0 2 | 0 1 | 8 9 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

| | | | |
|---|---|---|---|
| 20.402(b) | 20.405(c) | 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)

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| Philip J. North, Regulatory Compliance | 7 0 4 | 3 7 3 - 7 4 5 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 |
|---|---|---|---|---|---|---|---|---|---|
| E | J A | I M O D | B 0 4 5 | Yes | E | S J | R L Y | 5 4 4 0 | Yes |
| E | B A | D C C | W 1 2 0 | Yes | | | | | |

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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


On January 2, 1989, at 1523 hours, Unit 1 experienced a reactor trip from 100% full power. Investigation revealed that the cause of the trip was personnel error due to the tripping of two Reactor Protective System (RPS) channels during the performance of a RPS calibration procedure. A contributing cause to the event was procedural incompleteness. At the time of the trip, several Integrated Control System (ICS) stations, including the Feedwater Masters, were in manual control. This condition contributed to an overfeed situation in the steam generators causing a trip of the Main Feedwater Pumps. However, no overcooling condition was experienced from this overfeed due to the high decay heat load. The immediate corrective action was to stabilize the unit at hot shutdown conditions. Other significant corrective actions included: the replacement of failed components in the ICS BTU Limit and Emergency Feedwater circuitry, the counseling of Control Room Operators involved in the incident and the counseling and administering of disciplinary actions to I&E Technicians involved in the incident.


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

## INTRODUCTION

On January 2, 1989, at 1523 hours, Unit 1 experienced a reactor trip from 100% full power. Investigation revealed that the cause of the trip was the tripping of two Reactor Protective System (RPS) [EIIS:JC] channels during the performance of a RPS calibration procedure. The tripping of the two RPS channels by the Instrument and Electrical (I&E) Technician performing the work was contrary to procedural guidance given by the procedure in use. Therefore, this event is classified as a personnel error, due to a failure to follow a procedure. A contributing cause to the event was procedural incompleteness. At the time of the trip, several Integrated Control System (ICS) [EIIS:JA] stations, including the Feedwater Masters, were in manual control. This condition contributed to an overfeed situation in the steam generators [EIIS:SG] causing a trip of the Main Feedwater [EIIS:SJ] Pumps. However, no overcooling condition was experienced from this overfeed due to the high decay heat load. Other post-trip anomalies were: the swap from "auto" to "manual" of the main feedwater block valves; the failure of the 'A' steam generator emergency feedwater [EIIS:BA] control valve (1FDW-315) to properly control steam generator level and the failure of the ICS BTU 'A' side limiter to run back feedwater after the trip.

The immediate corrective action was to stabilize the unit at hot shutdown conditions. Other significant corrective actions included: the replacement of failed components in the ICS BTU Limit and Emergency Feedwater circuitry, the counseling of Control Room Operators involved in the incident and the counseling and administering of disciplinary actions to I&E Technicians involved in the incident.

## BACKGROUND

The Reactor Protective System (RPS) provides a two out of four logic for tripping the reactor in the event that a predetermined safety parameter is exceeded. Technical Specification 4.1, Table 4.1-1, defines the frequency of testing required for the RPS. The required test frequency is monthly, and normally one of the four RPS channels is tested each week. At the time of this incident, RPS Channel 'A' contained dummy bistables installed in the Reactor Coolant System (RCS) [EIIS:AB] Pressure/Temperature and High Temperature modules due to a failed RTD. A dummy bistable is essentially a "hard wire" across the RPS module which prevents that module from tripping. The dummy bistables were installed due to a failed RTD which would have caused the two RPS modules mentioned above to remain in the tripped state. With dummy bistables installed, the channel can remain in an untripped state with other modules in the RPS channel capable of performing their trip function. Regarding dummy bistables, Technical Specifications allow

only one RPS channel to be in manual bypass or contain dummy bistables at any one time. Manual bypass is a key activated state which bypasses all trip signals for that RPS channel.

The Integrated Control System (ICS) balances reactor power, feedwater flow and steam flow to produce a demanded megawatt output. During RPS Channel calibration and testing the Reactor and Feedwater ICS control stations are placed in manual. The BTU limiter associated with the ICS feedwater system functions to attempt to maintain a specified amount of degrees superheat in the steam exiting the steam generator. Should steam generator pressure increase, T(hot) decrease, final feedwater temperature decrease or RCS flow decrease to a point such that ICS demand becomes greater than the calculated BTU Limit, then the BTU Limit decreases the feedwater demand signal to reduce feedwater flow. On a unit trip, the BTU Limit, responding to the increase in steam generator pressure, would be expected to cause a corresponding decrease in feedwater flow. Due to the location in the electrical circuitry at which the BTU Limit modifies feedwater demand, this modification of feedwater flow should occur regardless of the position of the ICS Feedwater Master Control stations. These BTU Limits are bypassed when reactor power is greater than 25% of full power.

DESCRIPTION OF OCCURRENCE

On January 2, 1989, with Unit 1 at 100% power, the calibration/testing of the Reactor Protective System (RPS) Channel 'D' was in progress. This testing was being performed per IP/1/A/305/3 (Nuclear Instrument and Reactor Protective System RP Channel Calibration and Functional Test) and IP/1/A/305/3D (Instrument Procedure Data Package for RPS Channel 'D' Calibration and Functional Test). These procedures are used concurrently as IP/1/A/305/3 contains the written instructions while IP/1/A/305/3D contains the data sheets.

At 1410 hours, Instrument and Electrical (I&E) Technician 'A' tripped RPS Channel 'A' as required by procedure. At 1459 hours, I&E Technician 'A' placed RPS Channel 'D' in manual bypass. The technician was required, by procedure, to place RPS Channel 'A' in a tripped state prior to placing RPS Channel 'D' in manual bypass because Channel 'A' contained dummy bistables making it technically inoperable. Technical Specifications allow only one inoperable channel at any time, and if a channel contains a dummy bistable or is in manual bypass, it is considered technically inoperable.

At 1522 hours, Control Room Operator (CRO) 'A' placed the following Integrated Control System (ICS) stations in "manual": Steam Generator/Reactor, Delta T(cold), Loop 'A' and 'B' Feedwater Masters,

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Reactor and Control Rod Drive. Per the I&E procedure, only the Control Rod Drive station is required to be placed in "manual" for the test. However, training instructs the operators to place these other stations in "manual" to ensure adequate control of feedwater flow is maintained.

At 1523:36 hours, I&E Technician 'A' tripped RPS Channel 'D' causing a reactor trip. Channel 'A' was already in a tripped state from an earlier portion of the I&E procedure and should have been placed in manual bypass prior to tripping Channel 'D'. The Technician's action satisfied the two out of four logic necessary to cause a reactor trip. The two procedures from which I&E Technician 'A' was working were not in agreement. The instructional procedure states: "If one of the other channels is tripped, place the tripped channel in manual bypass. Document on additional sheet." However, the data procedure also has instructional guidance but omits the cautionary guidance. I&E Technician 'A' stated that he read the applicable steps in IP/1/A/305/3, then laid it down and used IP/1/A/305/3D to perform the work. The technician failed to observe the guidance given in step 10.15.4 of IP/1/A/305/3, and did not notice the lights which indicated that RPS Channel 'A' was currently in a tripped state. Consequently, when I&E Technician 'A' tripped Channel 'D', a reactor trip occurred.

At 1523:36 hours, the Unit 1 reactor tripped on RPS Channels 'A' and 'D'. All control rod drive breakers opened satisfactorily and all control rods fully inserted into the core. Following the reactor trip, Reactor Coolant System (RCS) letdown was isolated and RCS makeup commenced per the Emergency Operating Procedure.

At the time of the reactor trip, the ICS Feedwater Master stations were in "manual". CRO's 'A' and 'B' failed to run Feedwater (FDW) demand back manually or place the ICS control stations back into "auto". This led to an eventual overfeed situation on Steam Generator (SG) 'A'. Contributing to the overfeed condition on SG 'A' was the failure of the ICS BTU Limit on the 'A' train. On a reactor trip this limiter would be expected to decrease feedwater to the SG's no matter what the position of the ICS Feedwater Master stations ("manual" or "auto"). However, due to the failure of a power supply card in the multiplier module for '1A' Feedwater BTU Limit, this limit did not actuate. The BTU Limit functioned properly on SG 'B' and prevented a SG 'B' overfeed. Also, due to the high FDW demand signal and the close proximity of the ICS SG high level limit to the high SG level Main FDW Pump trip setpoint, the SG high level limit was unable to modify 'A' train FDW demand fast enough to prevent a trip of the Main FDW Pumps.

At 1524 hours, both Main FDW Pumps tripped on high level in SG 'A'.

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Emergency Feedwater (EFDW) actuated with both Motor Driven Emergency Feedwater Pumps (MDEFWP) and the Turbine Driven Emergency Feedwater Pump (TDEFWP) starting properly. Upon actuation of EFDW, CRO 'C' noticed that the 'A' SG EFDW Control Valve (1FDW-315) was not controlling properly. CRO 'C' took manual control of 1FDW-315 and closed it to prevent further overfeeding of SG 'A'. It was later discovered that a bad driver card in the Emergency Steam Generator Level Train 'A' Control Cabinet caused the malfunction of 1FDW-315. Upon verification of proper operation and flow from the MDEFWP's, the TDEFWP was secured per procedure.

After securing the TDEFWP, the Control Room Operators secured RCS makeup and re-established RCS letdown. Also, the '1A' High Pressure Injection [EIIS:BQ] Pump (which automatically started on the reactor trip) was secured. At this point, the CRO's noticed that the Main FDW Block Valves (1FDW-31 and 1FDW-40) were out of "auto" and were "open". These valves were in "auto" prior to the trip and they should have remained in "auto" and gone closed when the reactor trip occurred. Later investigation revealed that the brief power loss (which occurred when the unit auxiliaries transferred from the Main Transformer to the Startup Transformer) to the momentary contact switch for the Main FDW Block Valves caused the loss of a seal-in circuit relay (1RXB). The loss of this relay allowed the Main FDW Block Valves control to come out of "auto". Upon discovery of these valves being out of "auto" and out of their required position, the CRO properly positioned these valves to "closed".

At 1617 hours, the CRO lowered 'A' side Main Steam pressure to approximately 960 psig in order to reseat a Main Steam Relief Valve (MSRV). The CRO had received information from the field that one MSRV had not reseated after the trip. After reseating the MSRV, the CRO later raised the Main Steam pressure back to approximately 990 psig. Again the MSRV lifted and the CRO adjusted the Turbine Header Pressure back to approximately 960 psig to reseat the MSRV. At 1627 hours, the 'A' Main FDW Pump was restarted and EFDW secured.

The post trip response for reactivity, RCS pressure, average RCS temperature, pressurizer level and minimum SG pressures all remained within their expected ranges. The SG pressures ranged up to 1135 psig for SG 'A' and 1144 psig for SG 'B' within less than ten (10) seconds after the trip of the Main Turbine. This is a normal response for this plant. This is attributed to the opening setpoints for the MSRV's (the majority of which are set at 1100 psig) and is not a result of any off-normal equipment performance.

CAUSE OF OCCURRENCE

The event which caused the Unit 1 reactor trip from 100% power was the tripping of Reactor Protective System (RPS) Channels 'A' and 'D' during the performance of on-line testing. The Instrument and Electrical (I&E) Technician failed to follow his instructions in the Nuclear Instrument and Reactor Protective System RP Channel Calibration and Functional Test procedure which stated in step 10.15.4 : "If one of the other channels is tripped, place the tripped channel in manual bypass..." The I&E Technician failed to recognize that RPS Channel 'A' was tripped although he had tripped it himself earlier and its tripped condition was indicated by a light on the panel on which he was working. Consequently, when he tripped RPS Channel 'D', a reactor trip was initiated. Therefore, the root cause of this event is classified as personnel error, due to a failure to follow a procedure.

A contributing cause to this event was the inconsistency between the two procedures in use. IP/1/A/305/3 gives instructions on the performance of the test/calibration while IP/1/A/305/3D is intended to be a data sheet. However, IP/1/A/305/3D also gives procedural guidance and the cautionary guidance given in step 10.15.4 of IP/1/A/305/3 (regarding placing tripped channels in manual bypass) does not appear in IP/1/A/305/3D. This leads to confusion, as both procedures contain instructional guidance and the guidance is inconsistent between the two.

After the reactor trip, the Integrated Control System (ICS) Feedwater Masters were in "manual". The Control Room Operators failed to manually run Feedwater back, or place the ICS stations back into "auto". This action was in direct conflict with simulator and classroom training which had been provided to the operators and contributed to the overfeed of Steam Generator 'A'. Simulator and classroom training instructs the Control Room Operator to accept the responsibility of manual control of the ICS when ICS stations are not in "auto".

Investigation revealed that although no positive identification of the Main Steam Relief Valve (MSRV) which had to be repeatedly reseated was made, it is believed to be 1MS-5 (this belief is based on an approximate location based on visual observation). 1MS-5 is the only MSRV on Unit 1 which has not been rebuilt to date.

A review of Licensee Event Reports (LER) occurring during the past year revealed one other reactor trip due to personnel error (reference LER 269/88-09). Therefore, this event is classified as a recurring event. A review of the corrective actions from LER 269/88-09 revealed that those corrective actions could not have prevented this event. The personnel

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error in LER 269/88-09 was due to a lack of attention to detail in contrast to a failure to follow procedure on this event.

According to NPRDS criteria, the failures of the BTU Limit module 4-6-9, the Backup Level Driver Card for Steam Generator 'A' and the relay 1RXB are reportable. The BTU Limit module 4-6-9 is manufactured by Bailey Controls and is model #6618210-1. The Backup Level Driver Card for Steam Generator 'A' is manufactured by Westinghouse Electric and is model #2837A16G03. The relay 1RXB is manufactured by Struthers Dunn and is model #219XPXT-120.

No personnel injuries occurred as a result of this incident. At the time of the reactor trip, Unit 1 was operating with a Steam Generator tube leak. The steam release to the public, via the Main Steam Relief Valves, resulted in a total radioactivity release of 0.62 curie. This corresponds to a maximum dose to the public of 9.6 E-4 millirem, which is below the applicable 10CFR50.73 criteria. Consequently, the health and safety of the public were not compromised.

CORRECTIVE ACTIONS

The immediate corrective action was to stabilize the unit at hot shutdown conditions.

Subsequent corrective actions were to:

Repair the 'A' Steam Generator BTU Limit prior to unit startup;

Repair 1FDW-315 "auto" control prior to startup;

Counsel Instrument and Electrical (I&E) Technician 'A' and other members of the shift involved in the incident on the necessity of following written instructions per Station Directive 2.2.1;

Administer disciplinary action to the I&E Technicians involved in the incident for failure to adhere to written procedures;

Counsel the Control Room Operators involved in this incident were to ensure they understood the necessity of maintaining proper control of Integrated Control System (ICS) stations when these stations are in "manual";

Add instructions to the Unit 2 and 3 Reactor Operator Turnover Sheets, to verify that the Main Feedwater Block Valves go closed on a reactor trip.

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Planned corrective actions are as follows:

I&E will review IP/1,2,3/A/305/3 (Nuclear Instrument and Reactor Protective System RP Channel Calibration and Functional Test) and IP/1,2,3/A/305/3A,3B,3C,3D (Instrument Procedure Data Package for RPS Channel Calibration) to determine if instructional guidance should be given in both procedures. After this review, these procedures will be revised as necessary to ensure appropriate cautionary and instructional guidance is given in both procedures, as required, to ensure proper task completion.

I&E will create IP/0/A/305/14 (RPS Control Rod Drive Breaker Trip and Timing Test) for the performance of the Control Rod Drive Breaker Trip Test. This procedure will contain appropriate instructional and cautionary guidance to prevent a reactor trip in situations where a dummy bistable(s) is installed in a Reactor Protective System channel.

The Projects group will process a Nuclear Station Modification which ensures that the Main Feedwater Block Valves do not come out of "auto" upon the transfer of power from the Main Transformer to the Startup Transformer.

I&E along with Operations will examine the need to have ICS stations in "manual" for the performance of IP/1,2,3/A/305/3. Procedure changes will be made to Operations and I&E procedures as required.

The Main Steam Relief Valve (1MS-5) will be rebuilt.

ANALYSIS OF OCCURRENCE

After the trip, the reactor was safely stabilized at hot shutdown by the operators. Evaluation of the post-trip response concluded that two secondary control parameters exceeded the preferred or expected response range, but did not constitute a significant concern.

Concerning reactivity control, all Control Rod Drive breakers opened and all control rods fully inserted maintaining subcriticality. The Reactor Coolant System (RCS) pressure ranged from a minimum of 1765 psig to a maximum of 2205 psig. The average RCS temperature ranged from the pre-trip value of 579 degrees F down to 547 degrees F. The cold leg temperature reached a minimum of 546 degrees F. Concerning RCS inventory control, pressurizer level remained on scale without using safety injection. The minimum level reached was 44 inches, which is above the Duke Power Company standard of 40 inches for normal pressurizer level control.

The Steam Generator (SG) pressures briefly increased to post-trip levels of 1135 psig in the 'A' SG and 1144 psig in the 'B' SG. This response did not exceed the ASME Code limit of 1155 psig. The minimum SG pressures were 975 in SG 'A' and 965 in SG 'B', which were above the preferred minimum value of 925 psig. Operators lowered pressure to reseat a Main Steam Relief Valve in Loop 'A'.

Due to the failure of the 'A' Loop BTU Limit power supply (which prevented the post-trip Feedwater runback for the 'A' SG) and the failure of operators to manually run back Feedwater demand or place the ICS Feedwater Master control stations in "auto", SG 'A' level increased to the high level limit which tripped both Main Feedwater Pumps. However, all Emergency Feedwater Pumps started properly, supplied water to the SG's as required and remained operable throughout this event. Although there was an excessive Feedwater flow condition to SG 'A', the effects were not significant in terms of the primary system parameters of pressure, temperature and pressurizer level, as noted by the above discussion. This is due to the fact that SG pressure (and hence saturation temperature) did not decrease significantly, and Main Feedwater was rapidly replaced with Emergency Feedwater. RCS pressures and temperatures remained within the recommended Pressure-Temperature window.

Following the trip, both Main Feedwater Block Valves transferred from automatic control to manual and remained open due to the seal-in relay losing power when the unit's electrical loads transferred from the main transformer to the startup transformer. This action had little effect on the post-trip response because Main Feedwater was terminated within 58 seconds of the trip. The Emergency Feedwater Control Valve (1FDW-315) did not automatically control properly due to a failed driver card in the Emergency Steam Generator Level Control Cabinet's Train 'A' backup level control circuits. The operator quickly noticed that the 1FDW-315 was not closed, with SG level being higher than the control setpoint, and took manual action to close it. Again, this response did not adversely affect the post-trip response of any primary system parameters.

The unplanned release of 0.62 curie of activity via the Main Steam Relief Valves and corresponding dose of 9.6 E-4 millirem to the public is within the limits of 10CFR50.73 criteria. Based on the above analysis coupled with the fact that no safety limits were exceeded as a result of this event, it is determined that the health and safety of the public were not affected.

Duke Power Company
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Vice President
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**DUKE POWER**

February 1, 1989

U.S. Nuclear Regulatory Commission
Document Control Desk
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Subject: **Oconee Nuclear Station**
**Docket Nos. 50-269, -270, -287**
**LER 269/89-01**

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report (LER) 269/89-01 concerning a Unit 1 reactor trip on January 2, 1989.

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PJN/ler4

Attachment

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