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April 18, 1994

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 269/93-10, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is a supplement to Licensee Event Report (LER) 269/93-10, concerning equipment failure which caused low steam generator water level resulting in a manual reactor trip. This supplement is provided to correct a typographical error.

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,

J W Hampton

J. W. Hampton
Vice President

/ftr

Attachment

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NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1): Oconee Nuclear Station, Unit 1

DOCKET NUMBER (2): 05000 269

PAGE (3): 1 OF 7

TITLE (4): Equipment Failure Causes Low Steam Generator Water Level Resulting In Manual Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
11	03	93	93	-- 10 --	01	04	18	94	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 13

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (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
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
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: L. V. Wilkie, Safety Review Manager

TELEPHONE NUMBER (Include Area Code): (803) 885-3518

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
F	JJ	XCV	P070	Yes					
F	JJ	V	S578	No					

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 15 single-spaced typewritten lines) (16)

On November 3, 1993, Unit 1 was operating at approximately 88% Full Power due to a feedwater heater drain pump being out of service. A manual reactor trip at 13% Full Power was initiated at 0100 hours after a plant transient. This transient resulted from a momentary circuit interruption due to a broken wire termination lug in the terminal box on Main Steam Stop Valve (MSV) 2 causing MSV 2 to partially close. The other three MSV's closed due to an interlock. The circuit interruption then cleared and MSV 2 began to open. The other three MSV's received a signal to reopen, however, only MSV 1 reopened. Two MSV's, associated with the A Main Steam line, remained closed due to stuck test solenoid valves. This condition resulted in only the B Steam Generator (SG) supplying steam to the Main Turbine and A SG isolated with its' Main Steam Relief Valves open. This ultimately lowered SG B pressure and water level to a point where the reactor was manually tripped by the Control Room Operator in accordance with Operations Management Procedures. Emergency Feedwater initiated to restore SG level and other systems responded as expected to stabilize the unit. The root cause of the event is Equipment Failure. Corrective actions include repairing the wire lug in the control cabinet, inspecting and cleaning the test solenoid valves, and evaluating the need to require a manual reactor trip on low SG pressure.

BACKGROUND

Each Oconee unit has two Steam Generators (SG) A and B, and a Main Steam (MS) line associated with each SG. There are no Main Steam Isolation Valves in the Oconee design. Each MS line (A and B) separates at the turbine header into two lines before entering the Main Steam Stop Valves (MSV). There are four MSV's (MSV 1 and 2 on the B line and MSV 3 and 4 on the A line) supplying the Main Turbine steam chest. The primary purpose of the MSV's is to completely shut off steam flow to the turbine. The turbine is rolled during startups using MSV 2 to slowly admit steam to the steam chest. Each MSV has an associated Control Valve to control the amount of main steam flow to the turbine header. The Control Valves are opened to admit steam to control the turbine speed. When MSV 2 has been opened to >/= 95% a limit switch sends a signal for MSV 1, 3 and 4 to open.

The Integrated Control System (ICS) [EIIS:JA] provides fully automatic control of reactor power, steam generation rate, and generated load by processing selected signals of measured plant parameters. The Unit Load Demand (ULD) subsystem is designed to provide the operator a means to communicate desired electrical output to the ICS. It also recognizes various limiting conditions of the plant and ensures operation within these limits. Within the ULD subsystem is the track mode of operation. This mode of operation occurs due to some abnormal condition, where an ICS subsystem cannot correctly respond to the ULD. In track, actual generated megawatts electrical becomes the demand for the ICS (ULD).

EVENT DESCRIPTION

On November 3, 1993, Oconee Unit 1 was operating at approximately 88% Full Power because one feedwater heater drain pump was out of service for repair. At approximately 0056 hours, with the Integrated Control System (ICS) in automatic, a Main Steam Pressure High/Low Statalarm was received in the Control Room. RO-A observed the main steam pressure mismatch. Operators observed electrical megawatts and reactor power decreasing and the A turbine bypass valves [EIIS:SO] open in automatic. RO-A placed the A turbine bypass valves in manual and attempted to close them, thinking the cause of the pressure decrease was that the valves had failed open. By design, the high SG outlet pressure signal maintained the turbine bypass valves open in their overpressure protection mode. Subsequently, RO-B recognized that the Main Steam Stop Valves (MSV's 3 and 4) were closed. The Emergency High/Low SG B level statalarm was received at 0100:17 hours. RO-A checked the B SG extended startup level and noted 0 inches on the primary indication. RO-A announced that B SG level was less than 15 inches and a Reactor trip was required per the Operations Management Procedures. With the reactor at approximately 13% Full Power, RO-B initiated the manual

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reactor trip at 0100:25 hours. The Control Rod Drive [EIIS:AA] breakers opened, and all full length control rods were inserted into the core, shutting down the reactor. The main turbine/generator tripped automatically in conjunction with the reactor trip.

Prior to the unit trip, as a result of decreased electrical output and the unit in track, feedwater (FDW) demand was being reduced. This resulted in the continued drop of the level in the SG's. Since all the steam to the turbine was being supplied by the B SG, the level was dropping faster than the level in the A SG. Following the trip, the Main Feedwater Pump (MFDWP) speed was regulated to maintain a constant pressure differential across the FDW control valve for the A SG. At the same time, the FDW control valve for the B SG was opening to recover level. This combined with the low pressure in the B SG resulted in high FDW flow to the B SG. With the MFDWP's unable to respond, a low MFDWP discharge pressure resulted. At 0100:35 hours, when the MFDWP discharge pressure reached 770 psig, both motor driven emergency feedwater pumps (MDEFWP) started. The Turbine Driven Emergency Feedwater Pump (TDEFWP) started at 0100:42 hours, but, due to a design that requires a 15 second duration of the start signal, the TDEFWP shut down at 0100:52 hours after the MFDWP discharge pressures recovered within 10 seconds.

Operators took other manual action per the Emergency Operating Procedure (EP/1/A/1800/001). As normally required after a reactor trip the operators started a second High Pressure Injection (HPI) [EIIS:CB] pump at 0101:10 hours and opened 1HP-26, HPI loop A Emergency Make-up Valve, to maintain Pressurizer level. At 0113:13 hours, the operators closed 1HP-26 and stopped the second HPI pump.

Post trip parameters remained within acceptable limits. Reactor Coolant System (RCS) [EIIS:AB] pressure decreased from 2095 psig to 1790 psig then increased and controlled at 2135 psig. Pressurizer inventory varied from a high of 224 inches to a low of 66.5 inches before controlling at approximately 140 inches. RCS temperatures converged smoothly to approximately 554 F after the Emergency Feedwater (EFW) [EIIS:BA] started supplying the SG's. SG A pressure decreased from 883 psig to 742 psig but recovered to reach a post trip high of 1119 psig before controlling at 1016 psig. SG B pressure had decreased to 112 psig at the time of the trip but recovered to 1016 psig. Main Steam Relief Valve's reseated within minimum reseat pressures.

Main Feedwater pump B was secured at 0140 hours, and both MDEFWP's were secured at 0149 hours, with the Main Feedwater pump A supplying the SG's.

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Investigation into the cause of the unit transient revealed that at approximately 0056 hours, Main Steam Stop Valve (MSV) 2 started moving closed and sent a close signal to MSV's 1, 3, and 4. Two stop valves in Main Steam (MS) line A (MSV 3 and 4) and one in line B (MSV 1) closed completely. Then MSV 2 (MS line B) returned to the full open position and sent an open signal to MSV's 1, 3, and 4. MSV 1 (MS line B) reopened fully but the two valves (MSV's 3 and 4) in MS line A remained closed.

Instrument and Electrical personnel began an investigation into the cause of the MSV closures. The position unit and the servo coil circuit on MSV 2 were checked. The servo valve was replaced when electrical current measurements through the loop were found to be out of range. After replacement, the loop current was checked and again found to be out of range. The inside of the terminal box on MSV 2 was then inspected and a terminal lug was found broken. The terminal lug was replaced and then the current checked satisfactory. The other three MSV's do not contain servo's but all the valves have test solenoids. The two test solenoids on MSV's 3 and 4 were found stuck open which keeps the MSV's closed. This prevented MSV 3 and 4 from re-opening when MSV 2 re-opened. All four MSV test solenoids were removed and inspected. A residue was found in the internals of MSV 3 and 4 test solenoids. The valves were cleaned and replaced. MSV stroke testing was performed with satisfactory results.

EFW flow in both EFW headers exceeded the continuous operating flow limit of 1098 gpm as stated in the EFW Design Basis Document. The limit is imposed to protect the SG tubes from the effects of flow vibration. The maximum flow indicated on transient monitor plots was 1185 gpm and 1150 gpm for EFW headers A and B respectively. The flow was above 1098 gpm for approximately 4 seconds on the A EFW header and approximately 1.5 seconds on the B EFW header. The elevated EFW flow rates were determined to be of no consequence since they did not occur for extended periods of time. The concern in that situation would be wear and flow-induced vibration of the SG tubes.

An evaluation of EFW response indicated that the 1B MDEFWP could have exceeded the design flow of 600 gpm during the time between the start of the MDEFWP's and the TDEFWP. This instantaneous high flow was due to the extremely low SG pressure in the B SG. To assure that no pump damage occurred due to the high flow rates, the 1B MDEFWP was tested per procedure PT/1/A/0600/13A, MDEFWP Test, with satisfactory results. Vibrations and flow rates were unchanged from the previous test.

B&W Nuclear Technology (BWNT) was consulted to evaluate the affect of this transient on the SG's. Based on this evaluation, BWNT concluded that this transient was bounded by previously analyzed transients therefore, the SG's were not adversely affected.

The Post Trip Review was completed, station management gave the approval to restart and the reactor was started up and went critical on November 4, 1993, at 0141 hours. The turbine/generator was returned to service at 0430 hours.

CONCLUSIONS

The root cause of this event is Equipment Failure due to the failure of the wire termination lug in the terminal box of Main Steam Stop Valve (MSV) 2. The cause of the failure of the wire lug is being investigated and will be metallurgically analyzed. No evidence could be identified where the wire lug or wiring had been replaced since the original installation. The servo on MSV 2 had been replaced, however, it has a plug connection and could be changed out without having to be disconnected.

The failure of the test solenoid valves on MSV 3 and 4 is attributed to inadequate maintenance. Even though the failure of these two solenoid valves resulted in MSV 3 and 4 remaining closed, the failure of the wire lug caused the initial closure. All the MSV's are exercised monthly and during the test, the solenoid valves are actuated. The test solenoid valves are not routinely disassembled for cleaning because of possibly contaminating the Electro Hydraulic Control [EIIS:JJ] fluid. A work history search revealed that problems have been encountered with the solenoid valves not opening properly (such as: ports on servo valve found misaligned, limit switch contacts dirty, and a relay board failure). These problems were identified during startup testing and resulted in the MSV's or Control Valves not opening or operating improperly. The protective function of the MSV's is to stop the steam flow to the turbine during an emergency or normal shutdown. There were no problems identified with the MSV's not closing when required. Problems like those identified in this event, associated with the test solenoids, have not previously been encountered.

A review of operating experience data for two years prior to this event identified no events related to failed equipment resulting in low SG level requiring a subsequent manual reactor trip. Therefore, this event is not considered recurring.

The MSV operators are NPRDS reportable components. The operators are Parker Hanifin model number - special. The test solenoid valves for MSV's 3 and 4 are Vickers model number F3DG4S4012A50.

There were no personnel injuries, radiation exposures, or releases of radioactive materials associated with this event.

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CORRECTIVE ACTIONS

Immediate

1. Operations personnel took appropriate actions per the Emergency Operating Procedure to bring the unit to stable conditions.

Subsequent

1. The wire connection lug was replaced in the Main Steam Stop Valve terminal box and the wire connections for all four valves were inspected.

2. The test solenoid valves for all four MSV's were inspected and cleaned.

3. The 1B motor driven emergency feedwater pump was tested satisfactorily.

Planned

1. Inspect all the Main Steam MSV's, Control Valves (CV), and Intermediate and Intercept Stop Valves (IV) wiring cabinets periodically.

2. Evaluate the need to periodically disassemble, clean or replace the test solenoid valves.

3. Evaluate the results obtained from the metallurgical testing and perform any required corrective actions.

4. Evaluate the need to require a manual reactor trip on low SG pressure.

5. Evaluate the need to train operators for this event on the simulator.

SAFETY ANALYSIS

The reactor was manually tripped at an indicated level of 15 inches in either Steam Generator, as directed by the Operations Management Procedure. The Reactor Coolant System (RCS) response remained within the nominal post trip limits.

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 1	05000 269	94	- 10 -	01	7 OF 7

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

If all four of the Main Steam Stop Valves had gone fully closed, the unit would have tripped immediately on an anticipatory trip.

If the manual trip had not been performed, an automatic trip would have occurred from the Reactor Protective System (RPS) trip setpoints on high RCS pressure or on low feedwater pump discharge pressure (loss of feedwater). Additionally, the Anticipated Transient Without Scram Mitigation Safety Actuation Circuit (AMSAC) actuation would have caused a turbine trip, (with a resulting anticipatory reactor trip if reactor power was still above 28%). Also, if steam generator level had remained at less than 21 inches for a total of 30 seconds, emergency feedwater would have actuated on dryout protection. Therefore, there are several methods to assure that the unit will be placed in a safe hot shutdown condition after this type of event.

There were no releases of radioactive materials, radiation exposures, or personnel injuries associated with this event. The health and safety of the public was not affected by this event.