

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Oconee Nuclear Station, Unit 3															DOCKET NUMBER (2) 0 5 0 0 0 2 8 7					PAGE (3) 1 OF 04		
TITLE (4) Reactor Trip Due to High Reactor Coolant System Pressure																						
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 7	2 3	8 5	8 5	0 0	1	0 0	0 8	2 2	8 5						0 5 0 0 0							
OPERATING MODE (9)				THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more of the following) (11)																		
POWER LEVEL (10) 0 7 4				20.402(b)		20.406(e)		<input checked="" type="checkbox"/>		50.73(a)(2)(iv)					73.71(b)							
				20.406(a)(1)(i)		50.36(e)(1)				50.73(a)(2)(v)					73.71(e)							
				20.406(a)(1)(ii)		50.36(e)(2)				50.73(a)(2)(vii)					OTHER (Specify in Abstract below and in Text, NRC Form 266A)							
				20.406(a)(1)(iii)		50.73(a)(2)(i)				50.73(a)(2)(viii)(A)												
				20.406(a)(1)(iv)		50.73(a)(2)(ii)				50.73(a)(2)(viii)(B)												
				20.406(a)(1)(v)		50.73(a)(2)(iii)				50.73(a)(2)(x)												
LICENSEE CONTACT FOR THIS LER (12)																						
NAME S. G. Godwin - Licensing										TELEPHONE NUMBER 7 0 4 3 7 3 - 2 3 6 2												
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																						
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPROS													
X	J,K	I,X,C	B 0 4 5	Y																		
SUPPLEMENTAL REPORT EXPECTED (14)										EXPECTED SUBMISSION DATE (15)			MONTH	DAY	YEAR							
YES (If yes, complete EXPECTED SUBMISSION DATE)										NO												

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

On July 23, 1985 at 0237 hours, Unit 3 tripped from 74% full power (FP) on high Reactor Coolant System (RCS) pressure during a secondary feedwater (FDW) transient. The FDW transient was a decrease in total FDW flow to the Steam Generators (SGs) because of an erroneous signal from a failed Integrated Control System (ICS) module.

The immediate corrective action was to stabilize the unit at hot shutdown conditions. The supplemental corrective actions investigated the cause of the transient, found a failed ICS module, and replaced the failed module.

The failed module's signal caused a FDW transient that resulted in high RCS pressure. This resulted in a Reactor Protective System (RPS) high RCS pressure trip.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	8 5	— 0 0 1	— 0 0	0 2	OF 0 4

TEXT (If more space is required, use additional NRC Form 365A's) (17)

Description of Occurrence:

On July 23, 1985, Unit 3 was operating at 74% FP with 3 of 4 Reactor Coolant Pumps (RCP) running. One RCP was secured because of low oil level indications. At 0237 hours, with the Integrated Control System (ICS) in AUTO, a Feedwater (FDW) transient occurred which started with a reduction in a FDW flow and subsequently caused an increase in B FDW flow. B FDW flow increased to the B BTU limit which initiated an alarm and limited additional increases in B FDW flow. Total FDW was being automatically decreased by the ICS Unit. Rapid control room operator action of switching FDW controls to MANUAL was ineffective in changing plant parameters. During the total FDW reduction, RCS average temperature and RCS pressure were increasing. The reactor tripped on high RCS pressure at 0237:15 hours.

When the unit tripped, the main steam relief valves (MSRV) lifted and properly reseated at approximately 1000 psi with turbine bypass system functioning automatically to control header pressure.

During the trip recovery, a high pressure valve was opened and a High Pressure Injection Pump (HPIP) was started manually to maintain RCS inventory and ensure sufficient RCP seal flow. Approximately 3½ minutes after starting the HPIP, the pump was no longer needed and secured.

A work request was written to investigate and repair the ICS for potential cause of the unit trip. At approximately 0430 hours, an erroneous signal being produced in the ICS was located. A Static Multiplier Module was test probed in the cabinet prior to removal and found to be producing an output signal of +10.000 V which was not correct for the given input voltages. The erroneous output signal from the module caused the ICS to reduce FDW flow on the 'A' side thereby causing the FDW transient that resulted in the unit trip. After the module was replaced and the ICS was checked to be operational, the unit was restarted. The reactor went critical at 1125 hours and the turbine was on-line at 1746 hours.

At approximately 0337 hours, during the recovery from the unit trip, the operators experienced problems with the Turbine Driven Emergency Feedwater Pump (TDEFWP) getting erroneous start signal. The TDEFWP was locked out at about 0335 hours and returned to AUTO at 0558 hours without further problems.

Cause of Occurrence:

The FDW flow transient was initiated by the ICS which was operating in the AUTO mode. The automatic functions of the ICS were operating to properly control FDW flow at the time the erroneous signal had emerged into the ICS circuitry. Control Room Operator actions were proper and timely in their pursuit to intercept the impending plant changes. The root cause of the unit trip was the FDW transient resulting from the erroneous output signal from and ICS module. This event is assigned a Category X, Other because the module failed.

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

The module was found to be producing the excessive voltage output signal by troubleshooting while the module was still installed in the ICS cabinet. The failed module was removed from the ICS cabinet and replaced with a spare module. The ICS was checked out for additional problems and none were found. The failed module had been placed in-service on July 17, 1985.

A review of past incidents reveals that several unit trips have been caused by the secondary system's affect upon the RPS's anticipatory reaction. But within these previous incidents, only three unit trips were caused by the ICS or ICS components.

Analysis of Occurrence:

The post-trip plant response was normal, except for a slight overfeed of the "B" steam generator and an erroneous starting signal of the turbine driven emergency feedwater pump.

The maximum reactor coolant temperature was approximately 600°F, before the reactor trip. The minimum RCS temperature following the trip was approximately 550°F, which is close to the expected target value of 555°F. The maximum cooldown rate of 50°F/hr was not exceeded and overcooling of the primary side after the trip did not occur. The slight overfeed of the steam generator B caused a reduction of the B Loop steam generator outlet temperature by only a few degrees.

Maximum pressurizer level was approximately 240 inches at the time of the reactor trip as a result of the overheating of the reactor coolant system. Minimum pressurizer level after the trip was 106 inches. The pressurizer level was gradually raised to its normal value of 220 inches. The 3HP-26 valve was manually opened and the second HPI pump 3A was manually started to ensure optional RCS inventory control.

Reactor Coolant System pressure peaked at 2288 psig. The minimum pressure during this event was approximately 1850 psig. Both values are expected responses. The RCS pressure was later controlled between 2100-2200 psig by pressurizer heaters and pressurizer spray.

The steam pressure dropped from a maximum of 1076 psig to a post-trip minimum of 959 psig. This drop was observed when 3A MFWP was secured, but pressure recovered shortly thereafter. All main steam relief valves reseated properly.

Steam generator level in steam generator 3A decreased from a maximum valve of 175 inches to a post-trip minimum of about 20 inches. It was later brought up to the normal post-trip value of 25 inches. The initial level in the SG 3B was 75 inches due to only one reactor coolant pump running in the loop. Following the reactor trip, the level was increased to a peak of about 125 inches. The level was later reduced to the normal post-trip value of 25 inches. Main feedwater was available during the entire transient. Emergency feedwater was not started, other than an erroneous starting of the turbine driven EFW pump. No emergency feedwater was injected, since the EFW control valves remained closed.

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The health and safety of the public were not affected by this incident.

Corrective Action:

The immediate corrective action ensured that the unit was stabilized in a hot shutdown condition. The ICS was investigated and a failed multiplies module was found. the failed module was replaced and the ICS was checked for proper operation.

DUKE POWER COMPANY

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

August 22, 1985

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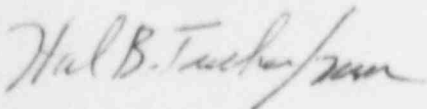
Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Oconee Nuclear Station, Unit 3
Docket Nos. 50-269, -270, -287
LER 287/85-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report 287/85-01 concerning a Unit 3 reactor trip due to high Reactor Coolant System Pressure. This report is submitted in accordance with 50.73(a)(2)(iv). This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,



Hal B. Tucker

SGG:slb

Attachment

cc: Dr. J. Nelson Grace, Regional Administrator
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