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FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
AUTH.NAME AUTHOR AFFILIATION
BENESOLE,S.G. Duke Power Co.
HAMPTON,J.W. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-011-01:on 911002,reactor trip results from electrical generator lockout after equipment failure in generator protective relay circuit.Investigation initiated to determine cause.W/920617 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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Duke Power Company
Oconee Nuclear Generation Department
P.O. Box 1439
Seneca, SC 29679

J.W. Hampton
Vice President
(803)885-3499 Office
(704)373-5222 Fax

DUKE POWER

June 17, 1992

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 269/91-11, Revision 1

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 269/91-11, Revision 1, concerning a reactor trip.

Further review of the loose connector problem and its planned corrective action (number 7) has revealed that training is a more effective method for resolving this problem.

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

Attachment

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, Georgia 30323

Mr. L. A. Wiens
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. P. E. Harmon
NRC Resident Inspector
Oconee Nuclear Station

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

M&M Nuclear Consultants
1221 Avenue of the Americas
New York, NY 10020

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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 WTH 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)
Oconee Nuclear Station, Unit 1	0 5 0 0 0 2 6 9	1 OF 0 9

TITLE (4) Reactor Trip Results From Electrical Generator Lockout After Equipment Failure In A Generator Protective Relay Circuit

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)
1 0	0 2	9 1	9 1	0 1 1	0 1	0 6	1 7	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 7 3

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)	X	OTHER (Specify in Abstract below and in Text, NRC Form 366A) 50.72(b)(2)(ii)
	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	
	AREA CODE	
S. G. Benesole, Oconee Safety Review Group	8 0 3	8 8 5 - 3 5 1 8

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
F	T L	C O N	G 0 8 0	X					

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)

On October 2, 1991, Unit 1 reactor tripped from 73 percent full power on a Reactor Protective System turbine anticipatory trip signal due to a generator lockout. The lockout occurred when a protective relay circuit spuriously actuated. Investigation found that the relay circuitry was interrupted when either a terminal wire or a connector became loose. The root cause was assigned equipment failure in the 41MXa circuitry. Post-trip operator response stabilized the plant. After the trip, the Emergency Feedwater System was actuated due to low feedwater pump discharge pressure during a condensate-feedwater pressure oscillation. One main feedwater pump was manually secured. The second main feedwater pump tripped due to high pressure at the pump discharge. Corrective actions include tightening connections, evaluating the need for a preventive maintenance program on electrical connectors near vibrating equipment, and investigation into low demand feedwater oscillations.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

BACKGROUND

The electrical generator [EIIS:GEN] uses protective relaying to sense abnormal conditions which may adversely affect the generator. When such a condition is sensed, it produces a lockout which will open the main generator breakers [EIIS:BRK], Power Circuit Breakers (PCB)-20 and -21, and trip the turbine. One of the protective circuits used to produce these actions is the relay circuit associated with the 41MXa relay [EIIS:RLY]. This relay senses the position of the main generator field breaker. If it is actuated while PCBs -20 and -21 are closed, a lockout relay (86GA) will actuate and the generator lockout will occur. Some of the wire terminals for this circuitry are mounted to the generator exciter housing. This housing is physically removed during some refueling outages. To facilitate its removal, the incoming and outgoing wires are fitted with connectors.

A turbine trip will produce a reactor trip when power is greater than 30 percent power by actuating Reactor Protective System [EIIS:JC] turbine anticipatory trip channels. The purpose of this trip is to limit Reactor Coolant System [EIIS:AB] pressure and prevent challenging the Power Operated Relief Valve.

The Emergency Feedwater [EIIS:BA] system will actuate on loss of both main feedwater pumps. The actual initiating conditions are a low discharge pressure (800 psig) on both main feedwater pumps or loss of hydraulic oil pressure on both main feedwater pumps. Main feedwater pumps will trip on high discharge pressure.

The control of feedwater while the two main feedwater pumps are operating is through the integrated control system [EIIS:JA]. The demand signal which determines pump speed is derived from the feedwater master demands and a signal proportional to the pressure difference across the feedwater startup valves, which control steam generator [EIIS:HX] level. Pump speed is determined by the position of control valves to the feedwater pump turbine. At normal speeds, low pressure steam is admitted and at high demand, high pressure steam is also admitted.

EVENT DESCRIPTION

On October 2, 1991, Unit 1 was operating at 73 percent full power. Startup from a refueling outage had occurred on September 29, 1991. The unit was being held at 73 percent full power for intermediate power escalation testing. The Turbine Driven Emergency Feedwater Pump (TDEFDWP) was inoperable while testing in the recirculation mode was being performed.

At 1555:27 hours an electrical generator lockout occurred. Immediately following this lockout the main turbine tripped. Since reactor power was greater than 30 percent, a Reactor Protective System (RPS) anticipatory signal tripped the reactor.

All full length control rods [EIIS:ROD] fully inserted into the core and the reactor was shutdown. One of two intermediate range neutron detectors [EIIS:IG], NI-3, did not return to its normal post-trip value. It decreased to approximately 5E-10 Amps and stabilized. All five power range detectors decreased normally following the trip. The two source range detectors came on scale and read approximately 200 to 300 counts per second, which are also normal post-trip values.

Following the reactor trip, the Reactor Coolant System (RCS) average temperature decreased from 580 degrees F to approximately 555 degrees F. RCS pressure decreased from approximately 2134 psig to 1886 psig forty seconds after the trip. Pressure then slowly increased to its 2155 psig setpoint. Pressurizer [EIIS:VSL] level reached a minimum of 86 inches one minute after the trip. Normal RCS letdown flow was first isolated and then reestablished to control pressurizer level above 100 inches. Level increased and stabilized at 114 inches. Steam generator pressures increased to a maximum of 1108 psig and then decreased to a minimum of 972 psig on the B steam generator and 995 psig on the A steam generator before leveling off at approximately 1018 psig. Steam generator levels decreased to a minimum of 21 inches on steam generator A and 20 inches on steam generator B before the 25 inch post-trip setpoint was maintained.

A feedwater oscillation occurred for twelve minutes after the trip. The two operating main feedwater pump (MFDWP) discharge pressures were oscillating with approximately 50 to 125 psig differences about their mean value of 1084 psig (1A MFDWP) and 963 psig (1B MFDWP). The frequency of these oscillations was approximately 5.6 cycles per minute. The oscillations in the two pumps were in phase, with both pump discharge pressures and speeds reaching maxima at the same time. After Reactor Operator (RO) A placed the B MFDWP in manual at 1605:34 hours, the oscillations continued but were then out of phase by 180 degrees. When 1A MFDWP discharge pressure was at a minimum, 1B MFDWP discharge pressure was at a maximum. Feedwater pump speeds and startup valve pressure differentials also followed similar oscillations. This cycling was divergent with the amplitude of each cycle increasing with time. While this was occurring, the minimum feedwater recirculation valve on the 1B MFDWP, which assures pump flow under shutoff conditions, was cycling between a closed and throttled position.

The 1B MFDWP was placed in manual by RO A and its speed decreased, with the intent of securing the 1B MFDWP. The operator is directed to do this by the Emergency Operating Procedure. At 1608:19 hours, the bypass valve for the condensate supply to the stator coolant coolers opened on a high pressure difference across the coolers. At 1608:25 hours the 1A and 1B Motor Driven Emergency Feedwater Pumps started, as did the 1B Condensate Booster Pump [EIIS:SD]. At 1608:33 hours the Condensate Demineralizer Bypass valve [EIIS:KD], 1C-14, opened. The 1B MFDW pump was tripped by RO A at 1608:35 hours. At 1608:50 hours, the ATWS Mitigation Safety Actuation Circuit (AMSAC), a backup emergency feedwater actuation system used to assure emergency feedwater under ATWS conditions, initiated. One second

later, the 1A main feedwater pump tripped. Transient monitor [EIIS:IQ] data was not available to determine the steam generator levels following these actuations. RO A has stated that steam generator levels controlled correctly at approximately 30 inches on the startup level following the emergency feedwater actuation.

Power Delivery Group personnel investigated the relay circuitry associated with the generator lockout. They found that relay 41MXa, which senses the main generator field breaker closure, had actuated but that the main field breaker had not opened at the same time. They also found the main generator lockout relay, 86GA, had also actuated. Further investigation found that the circuitry to the 41MXa relay had three loose connections where wires were connected to a terminal block (terminals EHC4, EHC6, and EHJ10). These terminal blocks are mounted inside the generator exciter housing cabinet. There are also connectors from a wiring harness to the bottom of the terminal cabinet. Three loose and unlocked connectors, EHC, EHF, and EHJ, were found. Circuitry drawings show that either the loose EHC connector or the loose terminal at EHC6 would have deenergized the 41MXa relay. The relay fails in a position which indicates an open main generator field breaker when it is deenergized.

Approximately one half hour after the reactor trip, while the main turbine was coasting down, eccentricity and thrusting of the generator exciter [EIIS:EL] shaft was noticed as well as noise from the exciter housing. It was found that the coupling between the exciter and the main turbine shaft was installed with one shipping bolt in place. Maintenance Engineering was asked to evaluate this condition. After conferring with General Electric personnel, it was decided that the thrusting was normal and could be expected. The noise was attributed to the exciter brushes and stopped when the brushes were pulled.

A Maintenance Investigation Report was initiated to determine the cause of the shipping bolt being left on the exciter coupling. This investigation determined that the coupling between the generator and the exciter had been changed during the previous refueling outage from a rigid type connection to a flexible type. Similar modifications had been made to Units 2 and 3. The modification package used to install the coupler included a written installation procedure from General Electric. This procedure gave instructions to remove the shipping bolts. The procedure was inadvertently left out of the Unit 1 exempt change package. The technicians who installed the coupling did remove seven of the eight coupling bolts, but left one bolt remaining on the coupling. A review of the exciter coupling bearing data indicated that vibration was not excessively high during the time that the shipping bolt was installed. The shipping bolt was removed from the coupling following coastdown of the turbine.

An investigation of the 1A main feedwater pump indicated that the last valve on the low pressure steam [EIIS:JK] control to the pump turbine was out of adjustment. The valve was readjusted correctly. This valve would have admitted high pressure steam if the pump turbine was operating at high

speed. A Maintenance Investigation Report has also been initiated to determine the cause and corrective actions for the misadjustment of the pump turbine controls.

CONCLUSIONS

An investigation into the root cause of the generator lockout revealed that there were three loose terminal wires and three loose connectors on the terminal board containing the circuitry for relay 41MXa. Either the loose EHC connector or the loose connection at terminal EHC-6 could have resulted in the inadvertent loss of power to the 41MXa relay. The exciter housing was removed during the previous refueling outage for maintenance. This required that the connectors be disconnected and then reconnected during assembly. The connectors are a twist-to-lock type and not screw-type. The fact that they were manipulated during the past outage suggests that it was the connectors that were at fault. On the other hand, the Power Delivery Group personnel who investigated the terminal board felt that the terminal connection was at fault based on the degree to which it was loose.

A review of past events showed that a similar trip occurred on Unit 1 on December 2, 1984. The wire at EHC-6 was found to be loose at the connector, not at the terminal. The Exciter cabinet had been removed during the previous refueling outage. The unit had been placed on line on November 29, 1984 and had experienced high vibration at the exciter-generator coupling. This indicates a recurring problem with Unit 1. No records of similar trips on Unit 2 and 3 could be found. The corrective action for the 1984 event was to tighten the wire to the connector and to inspect similar connections on Units 2 and 3. These actions did not prevent the latest trip. Exciter removal is scheduled for every third refueling outage at Oconee. This is considered a recurring event.

The root cause of this event is considered equipment failure. Either the terminal connection, EHC-6, or the EHC wiring harness connector interrupted the 41MXa circuit. Since six loose connections (three connectors and three terminals) were discovered, it is likely that vibration played a part in the failure. This is also supported by the fact that circuit continuity existed when the relay was initially energized during startup. The Maintenance Engineer in charge of the main turbine has stated that even though the exciter coupling had a shipping bolt incorrectly left on the coupling, the vibration at the exciter-generator connection was not measured as excessively high during the startup and subsequent operation at power. Furthermore, the location of the terminal block is such that it would tend to respond to general turbine floor vibration rather than the vibration from the exciter itself. Since both this event and the previous event occurred following maintenance which required moving of the exciter housing, it is also likely that this activity makes the terminals and connectors more susceptible to failure.

The exact component that failed is unknown, but measures will be taken to prevent the recurrence of both possible failures. Instrument and Electrical (I&E) personnel will inspect terminals and connectors near other vibrating equipment. The results of these inspections will be used to determine the need to establish a preventive maintenance program on that equipment. The suitability of the connectors used will also be reviewed to determine if they are appropriate for the present application. Connecting and locking electrical multi-pin plug connectors is a simple "skill of the craft" task. To help prevent recurrence of loose plug connectors, it will be emphasized to the personnel responsible for the work in the EH Cabinets, the necessity to ensure that the plug connectors are properly locked upon completion of work. In addition to these actions to decrease the possibility of the event recurring, an analysis will be performed by Design Engineering to determine the feasibility of changing the logic of the 41MXa circuitry so that the relay will not fail in the actuated position on loss of power.

After the reactor trip, power range neutron detectors and one of two intermediate range detectors fell to their normal post-trip values. One intermediate range detector, NI-3, remained at approximately 5E-10 Amps, which is higher than expected. Since all other indications of reactivity indicated the successful shutdown of the reactor, the NI-3 response was considered invalid. Instrument and Electrical personnel considered the response to be indicative of a detector failure, but they felt that the instrument would respond properly above the 10E-10 range. NI-3 did respond in this manner during the subsequent restart of the unit. It has been decided to replace the NI-3 detector at the next outage of sufficient duration.

Response of the primary system to the trip was normal. Reactor Coolant System (RCS) inventory, RCS pressure, and RCS temperature were all maintained within the normal post-trip range. The immediate response of the secondary system was also normal. Both steam generator pressure and level were maintained at or near their proper setpoint.

An upset in condensate-feedwater control occurred subsequent to the trip. Feedwater pressure began oscillating with a divergent trend. When the 1B main feedwater pump speed was manually lowered, the remaining feedwater pump was unable to quickly compensate for the drop in feedwater pressure. Feedwater pressure began to decrease so that several actuation setpoints were reached: 1) the stator coolant cooler bypass valve in the condensate system opened, 2) the condensate demineralizer bypass valves opened, and 3) a spare condensate booster pump started. In addition, the emergency feedwater system was actuated due to both main feedwater pump discharge pressures reaching their 800 psig setpoint. Both Motor Driven Emergency Feedwater Pumps (MDEFDWP) started and began delivering water to the steam generators. The Turbine Driven Emergency Feedwater Pump (TDEFDWP) was being tested. It started but was not aligned to the steam generators. It was secured rather than realigned because it was not needed. This action was consistent with the abnormal procedure for loss of main feedwater.

The effects of these actions was to compensate for the original pressure drop and increase feedwater pump discharge pressure. Steam generator level setpoint after emergency feedwater actuation was raised to 30 inches on the steam generator level. This shuts the normal startup control valves since they were controlling at 25 inches. The stator coolant and demineralizer bypass valve openings resulted in less pressure drop across the condensate system. The condensate booster pump start further increased pressure to the main feedwater pumps. The 1A main feedwater pump discharge pressure increased until the pump tripped at its high discharge pressure trip setpoint.

The cause of the feedwater swing and resultant loss of main feedwater is still under investigation. Several factors may have been contributing:

1. The feedwater pumps are equipped with minimum recirculation valves. These valves operate based on individual feedwater pump flow. The B MFDWP minimum recirculation valve was observed to be cycling open and closed during the transient. The control circuitry for these valves is being evaluated.

2. Because the unit had only had one effective full power day after an extended refueling outage and since the unit had only reached 73 percent full power, decay heat load was very low. This led to a low demand for feedwater following the trip. Normal Integrated Control System (ICS) control of feedwater at low demands may be unstable.

3. The control rack for the 1A MFDW pump turbine, which adjusts the amount of steam to the turbine, was found with the high pressure steam control valve out of adjustment. The main feedwater pump turbines can be supplied from either low pressure or high pressure steam. They normally operate, and are more stable, with low pressure steam. The significance of the poor adjustment is that the high pressure steam could have been admitted prematurely, thus reducing the degree of speed control on the pumps. The Maintenance Engineer in charge of these turbines states that this could have increased the feedwater oscillations but probably did not cause them.

Plant personnel are currently evaluating the various factors involved to determine the cause and potential corrective actions for this problem. A Maintenance Investigation Report is being prepared to determine the cause of the misadjustment of the 1A MFDWP controls.

The feedwater swing could have been terminated by placing both main feedwater pumps in manual mode. This would have prevented the pumps from attempting to follow startup feedwater valve pressure differences and stabilized pump speed demand. Simulator training will be evaluated to determine if post trip feedwater oscillations can be introduced to enhance operator training.

NRC FORM 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH 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.

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

This equipment failure is NPRDS reportable as the General Electric Alterex, model ATI. The event did not involve radioactive releases, overexposure to radiation, or personnel injuries.

CORRECTIVE ACTIONS

Immediate

1. Operations personnel safely controlled the reactor after the trip.

2. An immediate investigation was initiated to determine the cause of the reactor trip.

Subsequent

1. All loose terminal connections and electrical harness connectors found loose in the investigation were tightened.

2. The shipping bolt found on the exciter coupling was removed and rebalancing of the coupling performed.

3. After review, the Operations Emergency Operating Procedure on Unit 1 was changed to incorporate special directions required due to the failed neutron detector, NI-3.

4. Maintenance personnel readjusted the controls on the 1A MFDW pump.

Planned

1. Instrument and Electrical (I&E) personnel will inspect and correct as necessary terminals and connectors located on equipment subject to high vibration. Based on the findings of these inspections, an evaluation will be made to determine whether a preventive maintenance program to inspect these terminals on a periodic basis is required.

2. Engineering personnel will evaluate the failure logic of relay 41MXa to determine if alternative logic schemes are needed.

3. Maintenance Engineering will inspect the exciter-generator couplings for shipping bolts and remove any bolts, if necessary, on Units 2 and 3 during the next outage of sufficient duration.

4. During the next outage of sufficient duration on Unit 1, the detector for NI-3 will be replaced.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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

5. I&E engineers will evaluate the cause and effects of the feedwater swings which occurred at low power and make appropriate recommendations.

6. Operations Training personnel will evaluate the feasibility of introducing post-trip feedwater oscillations into operator simulator training.

7. Emphasize to applicable personnel, who are responsible for disconnecting and reconnecting plug connectors, the necessity for ensuring that the plug connectors, which have a means for locking, are properly locked upon completion of work.

SAFETY ANALYSIS

A generator lockout and the resulting turbine trip, while at power operation, leads to an imbalance between the amount of power produced in the primary system and the amount of power removed by the secondary system. This results in Reactor Coolant System (RCS) heatup and pressurization and an eventual challenge to the Power Operated Relief Valve (PORV). The Reactor Protective System (RPS) prevents excessive RCS overpressurization and heatup by use of the turbine anticipatory reactor trip. That is, the turbine trip will cause a reactor trip in anticipation of increased RCS pressure. This safety feature successfully actuated during this event.

The operator response to the trip was appropriate. However, the loss of main feedwater subsequent to the trip presented an unnecessary challenge to the Emergency Feedwater (EFDW) system. The EFDW system actuated and controlled steam generator levels appropriately. Although the Turbine Driven EFDW pump was being tested, it could have been aligned, had it been necessary, by the manipulation of one manual valve and two electric valves. If either Motor Driven Emergency Feedwater Pump had failed, one emergency feedwater pump would have been sufficient to remove decay heat.

The failure of intermediate range detector NI-3 to decrease below 5E-10 Amps resulted in one less indication of core reactivity. However, the existence of another valid detector with the same range, five power range detectors, and two source range detectors gave the control room operators sufficient alternate core reactivity indications.

No personnel injuries or radioactive releases occurred. The event did not compromise the health and safety of the public.