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FACIL:50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287

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SUBJECT: LER 90-002-00:on 900307,reactor trip caused by equipment failure,valve limit switch linkage became disconnected.
W/9 ltr.

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

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EXTERNAL:	EG&G STUART,V.A	4	4	L ST LOBBY WARD	1	1
	LPDR	1	1	NRC PDR	1	1
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DUKE POWER

April 5, 1990

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 287/90-02

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/90-02 concerning a reactor trip.

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,

H. B. Barron

H. B. Barron
Station Manager

RSM/ftr

Attachment

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	1 OF 1 3

TITLE (4)
Reactor Trip Caused by Equipment Failure, Valve Limit Switch Linkage Became Disconnected

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 3	0 7	9 0	9 0	0 0 2	0 0	0 4	0 5	9 0	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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)	[ ] 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: AREA CODE	
Henry R. Lowery, Chairman Oconee Safety Review Group	8 0 3	8 8 5 - 3 0 3 4

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
X	S J	X I S	F 1 3 0	Yes	X	A A	B R K	G 0 8 0	Yes
X	I G	X T	W 1 2 0	Yes	X	S J	X C V	L 2 0 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)

ABSTRACT

At 1406 hours on March 7, 1990, Oconee Unit 3 experienced a Reactor Trip from 100% Full Power due to high Reactor Coolant System (RCS) pressure. A valve limit switch linkage arm had become disconnected, probably due to vibration during start-up, causing a closed indication which satisfied part of the control logic for automatic closure of a feedwater block valve. The rest of the logic was satisfied when a routine test supplied a low reactor power input to the control logic. The block valve closed, resulting in a partial loss of feedwater and, subsequently the Reactor Protective System tripped on high RCS pressure. Trip response was normal except that one Control Rod Drive Breaker exceeded its expected trip time and was replaced. The limit switch linkage was reconnected and the unit was restarted. The root cause is identified as equipment failure.

BACKGROUND

The Feedwater (FDW) System [EIIS:SJ] at Oconee is divided into two headers, designated Loop A and Loop B, prior to entering the Steam Generators [EIIS:HX]. Each header consists of a Main FDW Control Valve [EIIS:V], a Main FDW Block Valve, a Start-up FDW Control Valve which by-passes the Main FDW Control and Block Valves, and other components which did not affect this event.

The valves can be controlled manually by the Control Room Operator (CRO) or automatically by the Integrated Control System (ICS) [EIIS:JA]. In automatic, the Main FDW Block Valve opens when the Start-up FDW Control Valve opens to 80% full open. If the Start-up FDW Control Valve closes to less than 50% open and the ICS neutron power indication is less than 10%, the Main FDW Block Valve receives a signal to close. The control circuit is such that, once the block valve receives a signal to either open or close, it must complete its travel before it can respond to a signal to reverse direction.

The Loop A Main FDW Block Valve, FDW-31, is a large motor operated valve which normally takes approximately 100 seconds to travel from full open to full closed.

The Loop A Start-up FDW Control Valve, FDW-35, is pneumatically operated and is located approximately 15 feet above floor level, above the main FDW flow header. Due to a multitude of pipes, valves, cable trays, etc., it is not readily visible from the floor. The limit switch [EIIS:XIS] on FDW-35 is a Fisher Controls Type 304 switch assembly which operates by having a linkage rotate an internal rod. The rod activates six switches which can be individually set at any point of valve travel. The linkage is inserted into a retaining clip attached to the valve stem. In order to follow the valve's motion, the linkage must be free to move in the clip as the valve moves, but should not be able to fall out of the slot.

The ICS receives its neutron power indication (NI) from either of two detectors located outside the reactor vessel [EIIS:VSL]. Normally NI-9 provides the ICS power signal and NI-5, 6, 7, and 8 provide the power signal to the Reactor Protective System (RCS) [EIIS:JC], Channels A, B, C, and D respectively. The capability exists for the ICS to take its input signal from the NI-5 signal via the RPS Channel A signal output. This substitution can be selected manually by the CRO or automatically by the Smart Automatic Signal Selector (SASS) [EIIS:IMOD]. Since the ICS input from NI-5 is processed through the RPS, it is affected by routine activities such as RPS channel testing.

EVENT DESCRIPTION

In November 1989, Oconee Nuclear Station Unit 3 shut down for a scheduled refueling outage. During this outage, two related Nuclear Station Modifications (NSM) were performed which revised the circuitry for several secondary system valves.

Part of the scope of NSM 32546 was to add open/closed indicating lights [EIIS:XI] next to the Control Room switches for the Start-Up Feedwater (FDW) Control Valves (FDW-35 and FDW-40). These lights were wired to existing limit switch contacts. As part of the planning for this modification, an NSM Training Package was distributed to appropriate reviewers at the station. The Training Package listed ten valves which would have position indicating lights added. It then listed three of these valves which would have new limit switches added and specified that all of the other seven valves would use existing limit switches. The assigned individual in Operations used this information to prepare a short modification description for use in training the Operators on the effects of this NSM. This Operations training package did not mention new limit switches, but specified that the indicating light would operate based on valve position rather than valve demand. Operations management questioned the Projects Accountable Engineer about the design of the new indicating light circuit and, based on his reply, understood that the lights were operated by different limit switches from those used for the automatic function of FDW-31. In one sense this is true, in that the switches are independently wired and set. However, they share a common housing and actuation linkage.

A second modification, NSM 32804, included replacement of the control switches for several valves, including FDW-35 and 40. The switches permit the CRO to place the valves in OPEN, AUTO, or CLOSED. As part of the Post Modification Testing (PMT) for these NSMs, FDW-35 was cycled and the limit switches and indicating lights were verified to be operating properly. The associated work requests (WR) were signed off December 12, 1989.

In addition to the PMT associated with these NSMs, IP/0/B/0275/5M, "Feedwater Control Valves and Interlock Calibration", was performed by Instrument and Electrical (I&E) following routine disassembly and repair of the valve. As part of IP/0/B/0275/5M, the 50% and 80% limit switch settings were tested and demonstrated to be functioning properly. This work was also completed December 12, 1989.

Additionally, routine Performance valve stroke time tests were performed. The Main FDW Loop A Block Valve (FDW-31) was stroke tested December 9, 1989 and FDW-35 was stroke tested December 15. The Performance test on FDW-35 was performed by having a technician time the valve motion by visual observation. However, the technician who performed this test stated that she observed the stem motion from the side opposite the limit switch assembly and did not notice if the limit switch linkage was properly connected during her test.

On December 15, at 1100 hours, Unit 3 exceeded 200 degrees/300 psi, starting up from cold shutdown. The reactor was taken critical on December 18, 1989 and reached 15% power at 1900 hours on December 19, 1989.

At 2230 hours of December 19, WR 025372C was written because NI-9, the neutron detector used to indicate reactor power to the Integrated Control System (ICS), suddenly (and spuriously) indicated a power level drop of 5 percent. The Control Room Operators (CROs) switched ICS input to another detector, NI-5, which is normally used to provide power indication to RPS channel A.

Due to problems with the electrical generator [EIIS:GEN], the unit was placed in hot shut down on December 20, 1989. During this evolution the Loop A and B Main FDW Block Valves (FDW-31 and 40) were closed. Unit 3 resumed start-up on December 22. The generator was successfully placed on line at 0823 hours on December 23, 1989. Starting at 1800 hours, power was increased at a rate of 3 percent per hour.

On December 24, 1989, CROs noticed that the position indicating lights for FDW-35 were malfunctioning and showed the valve to be closed when valve demand, indicated flow, and steam generator level all indicated that the valve was really open. They initiated WR 25436C at 0503 hours. On the WR they identified the equipment location as the control room. Both the CRO who originated the WR and the Unit Assistant Shift Supervisor who approved it stated that they did not associate the control board indication for FDW-35 with the automatic function of FDW-31. They also stated that they made no attempt to visually inspect the valve itself for indications of a problem with the limit switches.

On December 28, the assigned planner visited the control room as part of the planning process for the work request. He states that he also made no attempt to visually inspect the valve and/or limit switches as part of the planning process.

On January 2, 1990, Operations night shift gave clearance for I&E to begin work on WR 25436C. However, I&E did not actually attempt to perform any work until January 4. Due to the delay they were required to get clearance again and Operations day shift decided that it was inappropriate to perform maintenance on FDW-35 at power and changed the WR priority to 5A (outage). The Operations Unit Shift Supervisor on duty at that time could not recall the exact reason for this decision, but stated that he would not have permitted FDW-35 to be cycled for troubleshooting at power due to the effect on flow through the start-up header. Again, the Unit Shift Supervisor did not associate the control board indication for FDW-35 with the automatic function of FDW-31.

From January 16 to 24, I&E monitored and trended the NI-9 indication. On January 19, 1990, Unit 3 tripped during testing of the control rod drive power supplies (reference LER 287/90-01). During this period, including the trip and restart, no abnormal response was indicated by NI-9. Therefore, NI-9 was declared operable and returned to service. On January 25 and 26, I&E performed IP/3/A/0305/03A, "Instrument Procedure Data Package For RPS Channel "A" Calibration and Functional Test," the RPS channel A monthly check, while NI-9 fed the ICS.

On February 7, 1990, I&E repeated IP/3/A/0305/03A, again while NI-9 supplied the power signal to ICS.

On February 11, at 2137 hours, an alarm indicating a mismatch between NI-5 and NI-9 was received in the Control Room. NI-9 was subsequently removed from service, and NI-5 was used for the ICS and power range chart indication.

On March 7, 1990, IP/3/A/0305/03A was scheduled again. It was included on the daily work list by the Operations Unit Manager and his assistants, all of whom were aware that this time it would be performed while NI-5 was suppling the power signal to ICS. They were also aware that the FDW-35 position indicating lights were malfunctioning, but did not realize that the same linkage activated the limit switches which controlled FDW-31.

Since NI-9 was out of service and NI-5 was suppling the power signal to ICS, the CROs took ICS into manual at 1400 hours as directed by IP/3/A/0305/03A. At 1402 hours, the I&E technicians began taking action per the procedure which resulted in an expected momentary zero power indication. This indication of zero power, along with the indication that FDW-35 was closed, satisfied the automatic control logic for FDW-31 to receive a close signal and it started to close at 1403:53 hours.

As FDW-31 began to close, it started reducing the feedwater flow through loop A to Steam Generator (SG) A. The CRO recognized the reduction in feedwater flow (but not its cause) and promptly increased both Loop A and Loop B demand. This caused a slight increase in Loop B flow, but had little effect on Loop A. Therefore the CRO took the feedwater loop A main and start-up valve controls to manual and increased demand at 1405:32 hours. This partially stabilized flow at about 80 percent of normal. Due to the feedwater flow decrease, SG A level and pressure decreased.

Reactor Coolant System (RCS) [EIIS:AB] temperature and pressure rose due to decreased heat transfer in SG A, and the pressurizer spray valve opened to try to reduce RCS pressure. As total steam flow decreased, the turbine control valves reacted to control Main Steam [EIIS:SB] header pressure and the number four valve throttled closed. In yet another attempt to increase feedwater flow, the CRO took FDW Pump Turbine (FWPT) A into hand at 1406:07 hours and increased demand.

In automatic the ICS would call for a reactor power reduction and insert control rods to reduce power in a situation like this, but, in this case the reactor was in manual and the CRO did not act to reduce reactor power.

At 1406:26 hours, approximately two and a half minutes into the transient, the reactor tripped from 100% full power due to high RCS pressure. The immediate response of the plant was normal for such a trip. The turbine and generator tripped, both 4kv and 7kv electrical power supplies transferred to the start-up source, the turbine stop valves closed, the main steam relief and turbine by-pass valves opened. All Control Rod Drive (CRD) [EIIS:AA] breakers [EIIS:BRK] opened and all control rods [EIIS:ROD] fell into the core. It was subsequently discovered during the post-trip analysis that one of the CRD AC breakers tripped in 88 milliseconds, 8 milliseconds slower than allowed.

At 1407 hours the CRO returned the Main And Start-up FDW Control Valves to AUTO. The valves then controlled steam generator levels adequately.

As the feedwater valves closed, FWPT A was not able to respond because it was still in manual, and it tripped on high discharge pressure at 1407:22 hours.

At 1407:27 hours, FDW-31 finally indicated closed. The indicated elapsed stroke time was 217 seconds compared to a normal stroke time of 101 seconds. The Performance Section has established a maximum stroke time of 165 seconds for this valve. It was noted that the alarm typer [EIIS:IQ] print out also indicated that FDW-40, the Loop B block valve, did not start to close until 1407:28 but indicated closed only 73 seconds later, much faster than normal. However, these discrepancies were not recognized until several days after Unit 3 had returned to power.

At 1407:38 hours, the CRO opened the High Pressure Injection (HPI) [EIIS:CB] Emergency Make-up Valve (HP-26) and started a second HPI pump to increase make-up to the RCS. This response is frequently taken after a trip in order to assure that pressurizer level is maintained on scale and to try to keep level above the pressurizer heater banks. The minimum pressurizer level during this event was 75 inches. At 1410 hours, the CRO closed HP-26 and secured the second HPI pump.

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

About 1418 hours, approximately 12 minutes following the trip, the CRO had to lower the turbine bypass valve pressure setpoint slightly in order to reseat main steam relief valve 3MS-16. The set point for that valve is 1050 psig and it reseated at 977 psig which is within the 10% reseat tolerance.

The unit was stabilized at hot shut down conditions and WR 26662C was written to have I&E investigate the cause of the trip. A review of the Alarm Typer and Transient Monitor data indicated that FDW-31 had started closed shortly after I&E had reduced the control signal from NI-5 to the ICS and prior to any other transient activity. Therefore, Operations personnel strongly suspected that the cause was the interlock between FDW-35 and FDW-31.

The assigned I&E technician climbed up to a position to inspect FDW-35 and observed that the limit switch linkage was disconnected from the valve stem. The technician concluded that the linkage must have vibrated loose, but he examined the nuts and set screws holding the linkage parts and determined that they were all tight. He reconnected the linkage back into place. The valve was then stroked and he verified that the linkage moved properly. He could find no apparent defect which might have caused the linkage to become disconnected. During this stroke, the CROs verified that the control room indicating lights accurately reflected the valve status.

Subsequently, I&E visually inspected the start-up FDW control valves on Loop B and the other Oconee units. These inspections found that double nuts holding linkage parts on two of the valves had loosened, but not sufficiently to allow them to disengage.

The CRD breaker was replaced under WR 26664C and the trip time of the replacement was satisfactorily tested prior to restarting Unit 3. Additional testing of the removed breaker was performed in an attempt to determine the cause for the slow actuation time. No defect was discovered. I&E staff concluded that the deviation from normal is so slight that the cause may be a defect too minor to discover. It is also possible that the deviation occurred in auxiliary relays used solely to provide the Events Recorder indication, and not in the CRD Breaker itself.

During the post-trip analysis it was noted that the RPS Channel C did not trip on high pressure. The bi-stable was checked and found to trip at 2345 psig, the proper setpoint. Further evaluation noted that Channels B and D tripped slightly prior to reaching the setpoint value, such that the Channel C setpoint was never reached. Therefore, the fact that Channel C did not trip was not a failure.

CONCLUSIONS

The root cause of the transient which led to this unit trip was equipment failure in that the limit switch linkage on FDW-35 became disconnected. The exact reason why the limit switch linkage became disengaged is unknown but presumed to be vibration during the Unit 3 start-up. This failure has been determined to be NPRDS reportable. The valve operator is a Fisher Controls series 476 pneumatic operated piston actuator, model number 476L-1-5-CC, size 130, equipped with a Series 304 switch assembly and mounting hardware. Both the maintenance procedure which governs reassembly of the valve and operator, and the vendor's recommended maintenance documentation were reviewed. These documents provide very little detail as to how the linkage is to be attached. The Instrument & Electrical procedure which sets the limit switch positions verifies that the switches properly indicate valve position, which implies that the follower linkage is installed to some degree, but no specific guidance or installation tolerances could be found which would assure proper assembly. Maintenance is to perform a review of historical data to evaluate the extent of similar problems with this type of limit switch and take appropriate action.

The slow operation of the Control Rod Drive breaker, a General Electric Model AK-24-25, is also NPRDS reportable.

Additionally, the failure of NI-9 forced the performance of the Reactor Protection System (RPS) monthly tests while NI-5 was supplying the Integrated Control System (ICS) power signal. The exact cause of the NI-9 failure cannot be known until it can be investigated and repaired during a future outage of sufficient duration. NI-9 is a Westinghouse model WL23636 neutron detector and is NPRDS reportable. However, this failure is considered a contributing factor and not a root cause.

The slow stroke time of FDW-31 indicated on the Alarm Typer is substantiated by the Transient Monitor indication of Loop A flow. Prior to the actual trip, loop A feedwater flow had somewhat stabilized at about 60% of normal loop flow, giving about 80% of total flow for both loops. The plot of flow indicates that FDW-31 stopped or slowed considerably while in a partially open position, then finally completed travel after the trip. I&E Support has proposed a theory that the valve torque switch activated, stopping the valve, then reset itself due to high vibration after the trip. The valve is normally stroke tested with little or no differential pressure, and it has never been tested using Motor Operated Valve Analysis Test System (MOVATS) or equivalent technology. The valve operator for FDW-31 is a Limitorque model SMB-4T-200, and is NPRDS reportable. This anomalous behavior was not a cause of this event.

This trip could have been avoided if appropriate personnel had recognized the problem prior to the initiation of the RPS monthly test. A solution would have been to place the control switch for FDW-31 in OPEN for the duration of the test rather than AUTO. However, the problem was not recognized for reasons described below.

Both Operations and I&E personnel recognized that the ICS power signal would go below 10% several times during performance of the instrument procedure. Appropriate Operations personnel knew that the position indication for FDW-35 was malfunctioning and was indicating less than 50% open. All licensed personnel are trained on the automatic interaction between FDW-35 and FDW-31 when those conditions exist simultaneously. However, no one anticipated the valve interaction in this event.

One contributing factor is that the Operator Training Lesson Plan for the Feedwater System erroneously stated that the interlocks for FDW-35 are based on the valve demand signal rather than actual valve position as indicated by the limit switches. While both the Unit Supervisor who approved the repair WR and the Unit Supervisor who approved the priority change stated that they did not think about this interaction at all when they made their decisions, it is possible that they might have made a connection if the training had specified that the limit switches provided the signal.

Another confusion factor which affected the decision process was produced by the terminology of Nuclear Station Modification 32546. The vendor manual, Design, and Projects personnel considered each individually set and wired limit switch to be separate. This outlook did not consider that the switches were operated by a common mechanical linkage. The Operations staff considered each switch assembly to be one limit switch, thought that the control room indicating lights operated independently of any previously existing indications, and expected that the "new" switches added by the modification were separately actuated. Therefore they thought no special action was necessary.

It should be noted that there is no indication that the modification actually caused the failure, and, without the modification, the Operations and I&E personnel investigating the trip would have had less indication that the limit switch linkage was involved, and trip diagnosis and recovery could have been delayed.

Ideally, the possibility of an interaction involving NI-5, FDW-31, and FDW-35 (and similar valves in Loop B) could have been recognized during review and approval of the RPS test procedure. The procedure recognized that normal operation would be affected if it was performed while NI-5 supplied the ICS power signal for whatever reason. Therefore, a limit and precaution specified that the Control Room Operators (CROs) must take the ICS into manual control for the duration of the test. It did not address any other actions required due to unusual unit or system status. However, the Nuclear Production Department Administrative Policy Manual For Nuclear Stations states that "Written procedures, however, cannot address all contingencies and ... Procedures , therefore, should contain a degree of flexibility appropriate to the activities for which each is applicable." It is not considered reasonable that this procedure should be expected to identify all possible combinations of deficiencies which might cause problems. Therefore, the RPS test procedure is not considered deficient. However, now that this specific interaction has been identified, the procedure will be revised to address it.

The CROs did not recognize during the initial transient that the Main Feedwater Block Valve, FDW-31, indicated that it was in the intermediate position. They could not have taken any action to reopen it before it reached the closed position, but they would have known the immediate cause of the transient. Also, they did not attempt to reduce reactor power to match the reduced feedwater supply. Prompt action might have prevented the trip. Operator training includes a scenario where FDW-31 fails closed while the ICS is in AUTO. During that scenario, the simulator trips because the control rods cannot move fast enough to reduce reactor power as fast as feedwater flow is reduced. In this trip, CRO actions, combined with an apparent failure of FDW-31 to completely close, resulted in FDW flow staying high enough that the unit might have stabilized if reactor power had been reduced to about 80%. These minor deficiencies occurred during the transient and trip response while the primary effort was to combat the immediate symptoms.

There have been five reactor trips at Oconee within the past two years where equipment failure was known or suspected to be the root cause. However, none of these events involved inoperable limit switches or nuclear instrumentation. Therefore, this event is non-recurring and none of the corrective actions from these previous trips could have prevented this event.

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

CORRECTIVE ACTIONS

Immediate

1. Operators brought Unit to stable hot shutdown conditions.

Subsequent

1. Instrument and Electrical (I&E) personnel found and repaired the disconnected limit switch linkage.

2. I&E replaced the slow CRD breaker and tested the replacement.

3. I&E checked the Reactor Protective System (RPS) Channel C high pressure bi-stable setpoint to verify proper operation.

4. Production Training Services has revised the Feedwater system lesson plan to indicate that the Main Feedwater Block Valves operate off Start-Up Feedwater Control Valve limit switches rather than off the Integrated Control System (ICS) demand signal.

5. Maintenance and Transmission Dept. personnel attempted to determine the cause of the CRD breaker slow response time.

Planned

1. Maintenance will perform a search of historical records to evaluate the extent of similar problems with valve linkages on similar valves in the secondary systems. Appropriate action will be taken based on the results of this search.

2. Maintenance will look at ICS and related components/systems to determine the components potentially affected by variations in NI indications during RPS testing, NI calibrations, etc. This will include the potential effect of Smart Automatic Signal Selector (SASS) signal selection during the activity. Maintenance will then review these effects with Operations and revise procedures where deemed appropriate.

3. Operations will participate in the review by Maintenance and will specifically consult with Design Engineering concerning the safety aspects of placing the Main Feedwater Block Valve switches in OPEN during RPS tests and/or NI calibrations, especially if NI-5 is selected as power signal to ICS. Procedures will be revised where deemed appropriate.

4. Maintenance will perform and/or coordinate valve signature analysis and other testing as necessary to assure that 3FDW-31 will stroke properly.

5. All licensed operators will review this incident report.

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

SAFETY ANALYSIS

The plant response to this event was relatively normal and as expected. The Reactor Protective System tripped the unit on Reactor Coolant System (RCS) high pressure at the required setpoint. The Control Room Operator (CRO) response maintained all parameters within nominal post-trip ranges. Specifically, RCS pressure peaked at 2311 psig, dropped to 1782 psig, and controlled at 2125 psig. Pressurizer level increased to 263 inches prior to the trip, dipped to 75 inches, and controlled at 122 inches. A second injection pump was manually started and run for approximately two and a half minutes to help keep the pressurizer on scale during the initial cooling transient as the RCS temperatures converged to the post-trip setpoint of 555 degrees. Steam generator "B" level increased slightly prior to the trip due to flow being diverted when FDW-31 closed, then both steam generator levels dropped to the post-trip setpoint of 25 inches. Steam pressure peaked at 1075 psig, then controlled at 965 psig.

One item that did not perform as expected was the Control Rod Drive (CRD) breaker which indicated slower than normal trip time. The indicated trip time includes the response time of additional relays in the event recorder circuit, therefore it is not certain that the CRD breaker was actually the slow component. For conservatism, Oconee has designated 80 milliseconds as the limit at which corrective action must be taken prior to restart. The replacement breaker tested acceptably.

In the event of total failure of the breaker to trip, the trip function would be performed by two DC breakers in the same power circuit which also receive RPS trip signals. In this event, the DC breakers tripped properly, assuring that the trip function was performed. Furthermore, the actual breaker trip time of the AC breaker was totally adequate provided the breaker was not allowed to degrade further.

FDW-31 apparently failed to close within the Performance stroke time limit of 165 seconds. Final Safety Analysis Report (FSAR) Section 15.13 discusses steam line breaks, which are identified as the most significant overcooling transients. According to the discussion, Oconee can successfully survive the worst case steam line break with out taking any credit for either CRO or ICS action. This means that FDW-31 does not have to operate at all in order to limit the consequences of an accident. The stroke time limit was supplied by Design Engineering and is based on a proposed revision to the Steam Generator tube allowable wall thickness criteria. Design Engineering has determined that a valve stroke time limit of 165 seconds for the main feedwater block valves would limit temperature gradients and resulting stresses sufficiently to avoid tube failures following a steam line break, even if the allowable tube wall thickness is decreased. Since the current criteria are still in effect, no tubes have been left in service based on the proposed revision. Therefore, failure to meet the stroke time limit does not have a nuclear safety significance.

The principle item of safety significance in this event, i.e. the lesson to be learned, is that two defects were known to exist but a significant operational relationship was not recognized, resulting in an unacceptable condition. This event resulted in a unit trip with no adverse effects.

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