FORD 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

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

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SUBJECT: LER 91-005-00:on 910401,spurious ATWS sys actuation occurred resulting in manual reactor trip.Caused by design deficiency & procedure deficiency.Defective circuit board replaced. W/910430 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 11
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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DUKE POWER

April 30, 1991

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 287/91-05

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/91-05 concerning a spurious ATWS system actuation.

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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IE22

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

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 0

TITLE (4) Design Deficiency and Procedure Deficiency Cause Spurious ATWS System Actuation Resulting in Manual Reactor Trip

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 4	0 1	9 1	9 1	— 0 0 5 —	0 0	0 5	0 1	9 1		0 5 0 0 0
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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)	0 8 7		
[ ] 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)*
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	50.72 (b)(2)(ii)
[ ] 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
F	J C	I M O D	S 3 4 5	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 April 1, 1991, at 0415 hours, one channel of the Unit 3 Diverse Scram System (DSS), an Anticipated Transient Without Scram (ATWS) mitigation system, spuriously actuated and both channels were bypassed. After some troubleshooting verified the input instrument was not at fault, the Operations Unit Supervisor attempted to return both channels to service, while the unit was operating at 87% Full Power. As he did so, at 1119 hours, spurious signals actuated both channels, resulting in three groups of control rods automatically dropping into the core. Per procedure, the control room operator manually tripped the reactor in response to the dropped rods. The trip response was normal and the unit was stabilized, with no safety system actuations either required or received. Troubleshooting discovered that one channel had actuated due to Equipment Failure of a bad circuit board (contributing cause) and the second actuated due to a Design Deficiency (root cause) which generated a high signal spike in the second channel when returning the first channel to service. A second root cause of Deficient Procedure was assigned due to less than adequate steps to reset the system. Corrective actions included replacing the bad circuit board, modifying the system, and enhancing procedures.

BACKGROUND

Unit 3 reactor core design has 69 control rods (EIIS: ROD) that are divided into groups. Groups 1 through 4 are safety groups, used to provide shutdown margin, and groups 5 through 7 are regulating groups, used to regulate power level. The Control Rod Drive (CRD) System (EIIS:66) controls the operation of the control rods. Each of the regulating groups (5 through 7) has its own regulating (normal) power supply (EIIS:JX). There is one auxiliary power supply which can be used to operate safety groups as needed and regulating rods in case of a loss of one of the regulating power supplies. For each group, a programmer accepts operational commands from the CRD System and converts them into low power outputs which control Silicon Controlled Rectifiers (SCRs) that sequentially energize the six phases of the CRD stator (EIIS:MO) causing rod movement. The trip function interrupts the two redundant 24 VDC power sources for the programmer. If the power output from the programmer goes to zero, no power is supplied to the motor drive windings and the rods are dropped into the core.

The Reactor Protective System (RPS) (EIIS:JC) is a safety related system which monitors parameters related to the safe operation of the plant. The RPS provides a two-out-of-four logic for tripping the reactor when a predetermined setpoint is exceeded. This is done via the reactor trip module relays which de-energize the control rod drive breakers (EIIS:BRK) and the SCR Control Relays, causing rod insertion.

The Diverse Scram System (DSS) was installed by Nuclear Station Modification (NSM) 22817 on Unit 2 during a refueling outage in October 1990 and NSM 32817 on Unit 3 during a refueling outage ending March 30, 1991. DSS is one of two systems which act to mitigate the consequences of a potential Anticipated Transient Without Scram (ATWS) event, and which are independent of the RPS as required by 10CFR 50.62. The ATWS Mitigation Safety Actuation Circuit (AMSAC) monitors main feedwater (EIIS:SJ) pump parameters and can trip the main turbine and start emergency feedwater (EIIS:BA) pumps in the event of loss of both main feedwater pumps. DSS monitors Reactor Coolant System (RCS) (EIIS:AC) pressure. If RCS pressure exceeds a setpoint as sensed by a channel, a Programmable Logic Controller (PLC) (EIIS:XC) activates that channel. If both channels are activated, DSS will interrupt power supplies to CRD Groups 5, 6, and 7 and the Auxiliary CRD power supply, which causes these rod groups to be inserted into the core. DSS also resets the Turbine Bypass Valves (EIIS:SO) setpoint, which raises the post-trip RCS temperature, adding more negative reactivity due to the moderator temperature coefficient. These two ATWS systems share PLCs and are connected such that operation of a channel BYPASS/ENABLE switch affects that channel of both AMSAC and DSS. One design criteria was that the system not revert to one of one logic during test or maintenance. Therefore, operation of the switch to the BYPASS position for one channel removes the system from service.

As part of the modification process, the NSM Technical Support Leader and the Accountable Engineer generate an Information and Training Package, which describes the modification. This package is distributed to designated individuals in the other station work groups who review the information and identify and implement necessary procedures or procedure changes.

EVENT DESCRIPTION

Duke Power Company installed the ATWS Mitigation Safety Actuation Circuit (AMSAC) and the Diverse Scram System (DSS), two Anticipated Transient Without Scram (ATWS) mitigation systems, as a Nuclear Station Modification (NSM). Information and Training Packages were issued prior to October, 1990, for the Unit 2 NSM, and on Jan. 4, 1991 for the Unit 3 NSM.

Operations staff personnel reviewed the Information and Training Package and prepared a series of procedure changes to revise several affected procedures. They also prepared Operator Training packages for routing to appropriate Operations personnel prior to startup of the affected units. The Operator Training package contains a system description, lists of new alarms and indicators, and a summary of changes to affected procedures. PT/3/A/600/01, "Periodic Instrument Surveillance," step 12.6.5 instructs the operator to bypass BOTH channels of AMSAC/DSS if any channel is inoperable or generates an invalid trip signal. Also, although several Operating Procedures instruct the user to "ENABLE" AMSAC/DSS, none of them contain specific instructions on which switches to use or how to reset an actuated channel.

The Operator Training package states that "Once a DSS trip occurs, the signal is 'Locked-in' until the initiating event clears and the Diamond TRIP RESET button is depressed." The Diamond TRIP RESET button is located in the control room and previously reset only the Control Rod Drive (CRD) breakers. After the modification, it also functioned to reset both channels of DSS. OP/3/A/1102/2, "Reactor Trip Recovery," does contain a step to press the Diamond TRIP RESET button, but that step remained worded such that it refers only to the function of resetting the CRD trip breakers and does not specifically describe that this process resets DSS.

On the morning of April 1, 1991, Unit 3 was operating at 73 % Full Power (FP). Power level was being held steady for core physics testing as part of initial power escalation following the refueling outage. At 0415 hours, the DSS Channel 1 actuated. The operators on the night shift confirmed that all Reactor Coolant System (RCS) pressure indications were below the DSS actuation setpoint and determined that the actuation was spurious. In accordance with PT/3/A/0600/01, "Periodic Instrument Surveillance," both DSS channels were placed in "Bypass," making the system inoperable. By licensee commitment, when one or both channels are inoperable, the system must be restored to service within seven days or a written report must be submitted to the NRC within the next 30 days. Work Request (WR) 32489C was written to initiate troubleshooting and repair. A decision was made to have the work performed on day shift rather than call out personnel from the Instrument and Electrical (I&E) crews with maintenance responsibility for the affected equipment.

At 0600 hours, physics testing at the 73 %FP plateau was completed, and power escalation resumed at the rate of approximately 3 %FP per hour.
At approximately 0700 hours, shift turnover occurred.

The I&E crew assigned maintenance responsibility for DSS was in an all day training class and was not immediately available. Therefore, the WR was sent to the crew that maintains the RCS pressure instrument which supplies the input to DSS Channel 1 via an isolation module. At approximately 0830 hours, the WR was given to Instrument and Electrical (I&E) Technician A who proceeded to check the RCS pressure instrument and isolation module output. He could find no indication of either a real high pressure or any instrument malfunction to cause a spurious high signal on the isolation module output. Technician A made no attempt to troubleshoot the DSS Programmable Logic Controller (PLC), on which he had not been trained. I&E Technician A consulted with the staff support engineer assigned to DSS, Nuclear Production Engineer (NPE) A. The next troubleshooting step would have been to check the DSS Channel 1 input processing circuitry and logic. However, because the I&E crew assigned to DSS was in training and troubleshooting would not continue for some time, and because the problem appeared to be an intermittent one, Technician A and NPE A recommended to the Unit 3 Unit Supervisor, Senior Reactor Operator (SRO) A, that DSS Channel 1 be returned to service and be monitored to see if additional spurious actuations occurred. SRO A asked about returning Channel 2 to service also. NPE A and SRO A discussed the situation and agreed that both channels could be returned to service since the system used two out of two logic. Therefore, they thought, a repeat of the spurious actuation of Channel 1 could not cause the system to actuate.

NPE A intended to accompany SRO A to the DSS system control panel, located on the floor above the control room, but SRO A was detained for a period of time by other duties, so NPE A left to resume his own duties. Therefore, SRO A was alone when he went to the DSS control panel to reset the system, shortly after 1100 hours. SRO A had read and signed the Operator Training package for the DSS modification on Unit 3 several days prior to this event, but he had not yet had occasion to operate the system. He did review PT/3/A/600/01, in an attempt to find specific instructions, but, as previously stated, that procedure did not contain guidance on how to reset DSS.

SRO A established communications with Control Reactor Operator (CRO) A, in the control room and operated the Channel 1 BYPASS/ENABLE switch (EIIS:XIS) to the ENABLE position. When enabled, the Channel 1 indicating lights and control room alarm (EIIS:ANN) became active, and indicated that the channel was actuated. Neither SRO A nor CRO A recalled from the training package that the Diamond TRIP RESET switch in the control room had been modified to reset DSS after a system actuation. SRO A returned the channel to BYPASS, then observed a local reset button, labeled "DSS TEST CH. 1 RESET." He again turned the switch to ENABLE and pressed the local reset button, clearing the channel and the alarms.

SRO A then operated the Channel 2 BYPASS/ENABLE switch to the ENABLE position at 1119:33. Channel 2 also indicated that it was actuated, so SRO A pressed the local "DSS TEST CH. 2 RESET" button, but the channel did not reset. The system design is such that the local reset button is only active if the opposite channel is in BYPASS, but the training packages and procedures did not include that fact. SRO A pressed the "DSS TEST CH. 2 RESET" button again, without results, and was in the process of reaching for the BYPASS/ENABLE switch to return to BYPASS, when, at 1119:43, Channel 1 again actuated spuriously.

With both channels actuated, the two out of two logic was satisfied, and the DSS system actuated. Control Rod Groups 5, 6, and 7 dropped into the core, initiating a power transient. Reactor power, which had reached approximately 87 %FP during escalation, dropped rapidly to approximately 6 %FP (as indicated by neutron instrumentation). The Integrated Control System (EIIS:JA) initiated a runback. RPS Channel C tripped on pressure/temperature ratio at 1119:50. CRO A, observing the alarms and control room indications of multiple dropped rods, manually tripped the reactor at 1119:54, approximately 11 seconds after the DSS actuation. This was in accordance with AP/3/A/1700/15, "Dropped Control Rods" abnormal procedure.

The immediate response of the plant was normal for such a trip. All CRD breakers opened and the remaining control rod groups (groups 1 through 4) dropped into the core. The turbine generator tripped, both 4kv and 7kv electrical power supplies (EIIS:EA) transferred to the start-up source, the turbine stop valves (EIIS:V) closed, the main steam relief and turbine by-pass valves opened.

The Primary system response was normal. Reactor Coolant System (RCS) pressure dropped from a normal 2155 psig to approximately 1830 psig as a result of the DSS actuated rod insertion, decreased to a low of approximately 1780 psig following the manual reactor trip, and then increased and controlled at approximately 2150 psig. Pressurizer level decreased from 224 to 144 inches prior to the trip and continued to decrease immediately after the trip. At 1122 hours, the CRO opened the High Pressure Injection (HPI) (EIIS:CB) Loop A Emergency Make-up Valve (HP-26) and started a second HPI pump to increase make-up to the RCS in order to keep Pressurizer level above the Pressurizer heater banks. The Pressurizer level during this time dipped to a minimum of 61 inches before increasing. At 1123 hours, the CRO closed HP-26 and secured the second HPI pump. Pressurizer level was then controlled at approximately 153 inches.

RCS Hot Leg temperatures decreased from 598 F to 569 F and RCS Cold Leg temperature decreased from 560 F to 555 F prior to the trip. Following the trip, RCS Hot Leg and Cold Leg temperature converged and stabilized at approximately 555 F.

Secondary response was also as expected. Main Steam pressure initially decreased from 874 psig to 860 psig as a result of the dropped Control Rod Groups (decreasing heat source), prior to the reactor trip. Following the trip, Main Steam pressure initially increased to 1024 psig. The Turbine Bypass Valves opened to lower Turbine Header pressure to its proper post trip value of 1010 psig.

Feedwater flow and corresponding Steam Generator levels responded appropriately. Feedwater flow initially decreased due to the runback signal called for by the Integrated Control System due to the dropped Control Rod Groups. Following the reactor trip, Feedwater flow decreased to control Steam Generator levels at a minimum level of 25 inches.

By 1130 hours the unit was at stable hot shutdown, and the transient was terminated.

Personnel on the I&E crew normally assigned to the AMSAC/DSS system were recalled from the training class in progress and began troubleshooting. Testing identified a problem with the analog input module (circuit board) on Channel 1. This module receives the analog pressure input signal from the circuit isolator and converts it into a digital signal for processing by the Programmable Logic Controller (PLC). The initial indication was that it was loose, so it was removed and reinserted. However, approximately five minutes later Channel 1 indicated that it had actuated again. This time the module was replaced. The system checked out properly with the replacement module in place. However, this defect still did not explain how Channel 2 had been actuated.

At 1818 hours, additional testing of the DSS was in progress. Unit 3 was at hot standby with Group 1 control rods withdrawn 50% per procedure as a standby source of negative reactivity. Control Rod Groups 2 through 7 were fully inserted into the core. In accordance with the troubleshooting plan, which had been discussed with management and the operators on duty, both channels of DSS were actuated. As expected, DSS interrupted power to the Group 5, 6, and 7 Control Rods. However, none of the personnel involved recognized that the Group 1 rods had been held in position by the Auxiliary power supply, which was also interrupted. Therefore, the Group 1 rods were inserted into the core, unexpectedly.

This second DSS actuation did not result in a transient since the unit was already subcritical. The DSS was reset, and the control rods were again withdrawn 50% as specified by the start-up procedure.

At this point in time it was apparent that some system interaction was causing Channel 2 to actuate occasionally when Channel 1 was enabled. The cause of this interaction was not understood but it did not reoccur if Channel 1 remained enabled after Channel 2 was reset. This was considered acceptable pending further testing. DSS was returned to service and Unit 3 was restarted and went critical at 0325, April 2, 1991.

A plan was established to monitor the DSS by connecting it to the Unit 3 Transient Monitor computer, but, on April 2, a hypothesis was suggested for the cause. Plans were made to test this hypothesis, and on April 3, 1991, DSS was taken out of service for more testing. This additional troubleshooting identified the source of the current spikes which had actuated Channel 2. When either channel was to be tested, the other channel was designed to be placed in BYPASS. The bypassed channel's outputs were inhibited in the PLC program to prevent a spurious actuation of the system. For testing purposes, a design was provided which used the opposite channel's bypass switch to interrupt the current loop which provides the tested channel's input signal from the pressure transmitter. It was discovered that, when the bypassed channel was restored to service, the isolator for the tested current loop caused a surge when the circuit was restored. This surge could produce a momentary signal which exceeded the actuation setpoint. If the PLC was not scanning that parameter at that moment, the surge went undetected since its duration was shorter than the 7 millisecond scan time of the PLC. If the surge occurred during the portion of time that the PLC was scanning that parameter, the channel would actuate and seal in. Furthermore, the channel actuation could occur and be sealed in on a channel which was itself in BYPASS. In BYPASS the status indicating lights were not active, therefore, the operator could unknowingly ENABLE the channel while it was in the actuated state.

Several wiring changes were implemented to change these circuits on Unit 3. The current loops were changed so that each is shorted out by a test switch during testing and removed from its relation to the opposite channel's bypass switch. This will make the signal increase from a zero reading rather than decrease from a high reading. Additionally, a new indicating light and supporting logic was added to each channel to indicate if the channel is in the actuated state prior to enabling the channel. It had been observed during post-trip troubleshooting that the original system installation did not wire up the "press-to-test" input for light bulb checks. The sockets were rewired to permit checks for burnt out indicating lights. Additional functional tests were performed to verify that the changes were effective. Unit 2 was subsequently also modified and tested. These changes will be incorporated into the Unit 1 system design prior to installation, currently scheduled for the next refueling outage.

During the investigation of this event, it was observed that, on August 30, 1989, Duke Power Design Engineering provided an "AMSAC/DSS Final Design Description" as part of a submittal to the NRC. Section 2.9 of that document stated "Administrative controls are provided which require placing an AMSAC or DSS channel in BYPASS in order to provide a control room alarm anytime maintenance, testing, or repair of a channel takes place. These same administrative controls also prohibit placing more than a single channel scheduled for work in BYPASS." It also stated "...a permanently installed BYPASS switch ... is designed such that only one channel of logic can be bypassed at a given time." The Design Engineering personnel who prepared this document stated that the intent was to have two switches (one per channel) with administrative rather than hardware controls to prohibit having both channels bypassed at the same time. However, they failed to include such a limitation in the NSM scope document or in the system descriptions transmitted as part of the NSM design package. Operations personnel prepared operating procedures based on the NSM design package information and were unaware of this commitment when procedures were revised. The operating procedures require placing both channels in BYPASS if one channel actuates spuriously.

CONCLUSIONS

It is concluded that this event was initiated by the actuation of DSS Channel 2 due to a signal spike induced by enabling Channel 1, followed by a spurious Channel 1 actuation caused by a defective circuit board on Channel 1. It is further concluded that this event could have been avoided if the Diamond TRIP RESET button, instead of the local test reset button, had been used to reset Channel 1 when it was observed to be in the actuated state after it was enabled.

One root cause of this event is Design Deficiency, Unanticipated Interaction of Components, due to Design Oversight. Design Engineering personnel did not foresee the current surge which could result in a channel being spuriously actuated during the return to service of the other channel. They also did not provide a status indication to allow the operator to verify that the channel was not actuated prior to return to service. These deficiencies created the possibility that the Diverse Scram System (DSS) could actuate anytime the two channels were placed in the ENABLE position.

An additional root cause is Deficient Procedure, Incomplete Information. Several Operating Procedures were revised to include the new DSS. However, none of these procedures contained any instructions on how to reset a DSS channel, if it was actuated. If appropriate procedures had given direction for the operators to press the Diamond TRIP RESET button to clear the observed Channel 1 actuation, it would have also cleared the Channel 2 actuation prior to Channel 2 being enabled. This would have prevented a DSS actuation, in this case, even if Channel 1 were subsequently re-actuated either by a spike when enabling Channel 2, or by the defective analog input module. It is recognized that similar failure scenarios could have resulted in DSS actuations in spite of improved procedures.

A contributing cause is Equipment Malfunction. The DSS actuation would not have occurred on this occasion without the spurious signal produced by a bad circuit board. However, the system was designed with the intent that a failure could occur without actuating the system. Therefore, this failure is not considered a root cause. The defective part was a Square D, "SY-MAX" Class 8030, Type RIM-125, 16 channel analog input module. This failure is NPRDS reportable.

An additional procedural deficiency was observed in that IP/3/B/0276/001, "ATWS Mitigation System AMSAC/DSS Functional Test," the I&E procedure for testing the DSS with the unit at shutdown, did not contain any cautions or other limits to prevent group 1 CRDs from being on auxiliary power while at 50% withdrawn during testing. This could result in an unanticipated insertion of group 1 rods as occurred in the second system actuation.

An additional design deficiency was observed for incomplete documentation. A restriction against placing both channels of AMSAC/DSS in bypass was described in the system description supplied to the NRC on August 30, 1989. Due to oversight, Design Engineering did not duplicate this restriction in modification documents transmitted to the station. Therefore, the restriction was not implemented. However, it is concluded that such a restriction is technically unnecessary because the entire system becomes inoperable when one channel is placed in BYPASS. Additionally, each channel has status lights which give continuous control room indication when the channel is in BYPASS.

This event is considered to be recurring. Two previous Unit 3 trips occurred due to equipment failure causing insertion of Control Rod groups into the core, followed by manual reactor trips, (reference LERs 287/90-01 and 287/90-03). However, the components involved and modes of failure were different in each case and, therefore, this event could not have been prevented by any corrective actions from those events.

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

CORRECTIVE ACTIONS

Immediate

1. After Diverse Scram System (DSS) actuation, the Operators manually tripped the reactor and brought the unit to stable hot shutdown conditions.

2. Instrument and Electrical (I&E) Technicians performed additional troubleshooting of DSS Channel 1. They identified and replaced a defective circuit board.

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Subsequent

1. An Operations Information Notice providing instructions for returning AMSAC/DSS to service was prepared and distributed prior to unit restart. This notice was referenced on the shift turnover sheets and was filed with the Unit Supervisor for ready reference by operators.

2. Troubleshooting identified the signal spike associated with restoring the transmitter current loop to service. The circuits were revised on Units 2 and 3 to eliminate the problem. Lights were also added to indicate if the channel is actuated while in BYPASS.

3. Unit 2 and 3 Operating Procedures were revised to provide adequate guidance on DSS operation to the operators.

Planned

1. The DSS revisions made to Unit 2 and 3 as a result of this event will be incorporated into the design prior to installation of the system on Unit 1.

2. In conjunction with installation of DSS on Unit 1, Unit 1 Operating Procedures will be revised to provide adequate guidance on DSS operation to the operators.

3. Testing will be performed on the defective circuit board to determine the cause of failure.

4. I&E will develop procedural guidance to specifically address troubleshooting DSS/AMSAC. Also, appropriate I&E procedures will be enhanced to adequately address the possibility of DSS actuations causing control rods on auxiliary or regulating power to drop during testing.

5. Design Engineering will propose additions to the Electrical Discipline Workplace Procedures Manual which will specifically address the review process for designs which use Programmable Logic Controllers and where analog/digital signal interfaces are utilized. This revision will specifically address the design consideration of potential signal interactions such as occurred in this event.

SAFETY ANALYSIS

The plant response to this event was normal and as expected. No Engineered Safeguards system or emergency feedwater actuations were either required or received.

Reactor Protective System (RPS) Channel C tripped on pressure/temperature ratio and the Control Room Operator (CRO) tripped the unit manually before a second RPS channel actuated. The CRO responses maintained all parameters within nominal post-trip values. Specifically, Reactor Coolant System (RCS) pressure dropped to a low of approximately 1780 psig following the trip, and then increased and controlled at 2150 psig. Pressurizer level dipped to a minimum of 61 inches before being controlled at approximately 153 inches. A

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

second High Pressure Injection pump was started and run for approximately one minute to help keep the pressurizer level on scale and above the pressurizer heaters. RCS temperatures converged and stabilized at approximately 555 F. Steam pressure increased to 1024 psig then the Turbine Bypass Valves opened to lower pressure to approximately 1010 psig. Feedwater flow decreased to control Steam Generator levels at a minimum level of 25 inches.

The Final Safety Analysis Report (FSAR) sections 4.5.3, 7.6, and 15.7 contain analysis of a single dropped rod. The Nuclear Engineering Support Section of Duke Power Design Engineering generally feels that the dropping of a group of rods, while not specifically analyzed, would make it very difficult for the unit to successfully runback to a lower power level and not trip. Reactor power tilt/imbalance related problems due to multiple dropped rods from one group should be less significant than the consequences of a single rod drop due to the distribution of the group rods in the core. A manual or automatic trip would terminate the initial transient and prevent the reactor from exceeding design parameters. Station Operating Procedures require the immediate manual trip of the reactor if more than one control rod drops. In this event, the operators diagnosed the event and manually tripped the reactor within 11 seconds. Post trip analysis indicates that the insertion of groups 5, 6, and 7 by DSS resulted in the reactor being subcritical by approximately 1 % delta k/k prior to the insertion of the safety groups.

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.