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Oconee Nuclear Station
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DUKE POWER

February 20, 1990

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 287/90-01

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/90-01 concerning control rod group drop during testing due to unknown reasons causes automatic reactor trip.

This report is being submitted in accordance with 10 CFR 50.72 (2)(ii). 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 6

TITLE (4) Control Rod Group Drop During Testing Due to Unknown Reasons Causes Automatic 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 1	1 9	9 0	9 0	0 0 1	0 0	0 2	2 0	9 0		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)	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(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
Henry R. Lowery, Chairman Oconee Safety Review Group	AREA CODE 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 1 S	M 2 2 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 January 19, 1990, Unit 3 tripped from 49% Full Power (FP) at 0849 hours following a rapid reduction in power from 100% FP after Control Rod Group 6 dropped into the core. A test was in progress at the time to verify proper operation of the control rod power supplies. During the testing of Group 6 power supply, transfer of power to the auxiliary power supply was initiated. The normal power supply apparently failed to disengage properly and later technician actions resulted in opposing phases being energized. The resulting opposing electromagnetic fields caused the rods in Group 6 to fall into the core. Operators realized the rod group had dropped from control room indicators, but were not able to manually trip the reactor before the Reactor Protective System automatically tripped the unit on low Reactor Coolant System pressure. In spite of several post-trip failures and abnormal problems the unit was stabilized at hot shutdown conditions. An immediate investigation was initiated to assess the causes and effects of the trip. The root cause of this event is classified as Unknown, Possible Equipment Malfunction. The major corrective actions were to provide more instructions in procedures to correct a contributing cause of Management Deficiency.

☆U.S.GPO:1984-0-454-481/18750

BACKGROUND

Unit 3 core [EIIS:AC] design has 69 control rods [EIIS:ROD] that are divided into eight groups. Groups 1 through 4 are the safeties and are in the full out position during normal power operation to provide safe shutdown capability. Groups 5 through 7 are the regulating groups and are used to control the reactor power during operation. Group 6 has up to 12 rods. Group 8 rods are the axial power shaping rods and are used to help control the power imbalance in the core within specified limits. The Control Rod Drive Mechanism [EIIS:66] associated with each control rod is driven by a variable reluctance motor. The stator windings are 6 phase star connected for operation in a pulse stepping mode. Each of the regulating groups (5 through 7) have their own regulating (normal) power supply [EIIS:JX]. There is one Auxiliary power supply which can be used to operate regulating rods as needed, in case of a loss of one of the regulating power supplies or to perform tests as was the case in this event.

Technical Specification 4.1, Table 4.1-2, "Minimum Equipment Test Frequency", requires a monthly Control Rod Movement test to be performed when the reactor is critical. This test is conducted by directions provided in Operations procedure, PT/0/A/600/15, "Control Rod Movement." However prior to performance of this procedure, it is normal station practice to conduct test procedure IP/3/B/0340/02, "Control Rod Drive DC Hold Supply, Regulate Supply, SCR Gate Drive, And Programmer Checks", to verify operability of the control rod power supplies. This verification test is conducted to ensure that no power supply problems exist that could cause a rod drop during actual movement of the rods.

IP/3/B/0340/02 requires the transfer of power for the rod group being tested from its normal power supply (regulating) to the auxiliary power supply. As a result of this transfer, the normal supply is disconnected from the control rods freeing the normal power supply for further operability testing. During this operability testing, the normal power supply is considered to be out of service.

The basic function of the Integrated Control System (ICS) [EIIS:JA] is to match the generated megawatts with the demand for megawatts. The ICS does this by coordinating the flow of steam to the turbine and the rate of steam production. When the ICS is placed in manual, reactor operators perform this function.

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 safety setpoint is exceeded. This is done via the reactor trip module relays [EIIS:RLY] which deenergize the control rod drive breakers and the SCR Control Relays, causing rod insertion.

EVENTS DESCRIPTION

On January 19, 1990, with Unit 3 at 100% Full Power, Standing Work Request 55741A was initiated to check the Unit 3 Control Rod Drive (CRD) power supplies. This check is normally performed prior to a monthly Control Rod Movement Test, required by Technical Specification 4.1. The purpose of the check is to verify proper operation of the Regulating Supply Gate Drives, Silicon Control Rectifiers (SCRs), Programmers, and Direct Current (DC) Hold Diodes to ensure no power supply problems exist that could cause a rod drop during the required test to follow.

After some initial pre-job discussion between Reactor Operators (RO) "A" & "B" and Instrument & Electrical (I&E) Technicians "A" & "B", the power supply checks were initiated under I&E procedure, IP/3/B/0340/02, "Control Rod Drive DC Hold Supply, Regulate Supply, SCR Gate Drive, And Programmer Checks." First, the I&E technicians tested the DC Hold Supply. Operations then placed the Integrated Control System (ICS) in manual so that the Auxiliary Power Supply could be tested. The Rod Group 5 Power Supply was then tested. Assistance was required from RO "A" & "B" to perform and verify several procedure steps. Instructions were given and steps were performed and verified via telephone conversations between the operators and the I&E technicians. During the test, a bulb was found burned-out on the Diamond Power Panel (see Attachment # 1) Clamp Release indicator light (in the control room). This was the only problem found and the bulb was replaced.

Shortly thereafter, the same test was performed on Group 6 Power Supply as required by procedure. Again, assistance was required from Operations personnel (ROs "A" and "B") to perform and verify certain steps of the procedure. Step 10.4.2.a required the transfer of control rods from Group 6 Regulating Supply to Auxiliary Supply. Step 10.4.2.b required the JOG-RUN switch on the Diamond Power Panel to be in the run position and step 10.4.2.c required the AUTO-MANUAL pushbutton to be in the manual position. Step 10.4.3 required Operations to verify that clamp release was indicated on the Diamond Power Panel. At this point, I&E Tech "A" contacted RO "A" by telephone to ensure that the above conditions had been established.

The proper test conditions had been established by RO "B", who had earlier performed the required steps needed for the test by directions from a parallel Operations procedure, OP/0/A/1105/09, enclosure 4.2, "Transfer Rods Between Normal And Auxiliary Power Supply." RO "A", upon visual observation of the Diamond Power Panel, verified the required test conditions and informed I&E Tech "A".

RO "A" stated during this report investigation that the Clamp Release lamp was lit on the control panel during his observation and prior to verifying the test conditions to I&E Technician "A". This procedure step verifies that the normal power supply has been disengaged and that transfer has been made to the auxiliary power supply. However, during this test, the normal power supply apparently failed to disengage properly. As a result of this failure, when I&E Technicians "A" and "B" cycled the normal power supply to the various phases (as required in later procedure steps), four phases of the Group 6 rods were energized instead of the normal two phases (see attachment # 2). When the technicians noticed this fact (from 4 Motor Power Signal Assembly phase lamps) and realized this condition was abnormal, they decided to cycle the supply back to its starting position and investigate. No procedure direction was given for this action. In the course of cycling the power supply, four phases became energized. The resulting opposing electromagnetic fields caused the Control Rod Drive Mechanism to release and the rods in Group 6 to gravity fall into the reactor core.

Immediately after the rod group drop, at approximately 0848 hours, reactor power dropped to 38% Full Power; then, due to a temperature decrease in the Reactor Coolant System (RCS) [EIIS:AB] and a negative temperature coefficient, power increased to 49% power.

At 0849 hours, approximately 18 seconds after the rod group dropped, the reactor [EIIS:RCT] automatically tripped on low RCS pressure of approximately 1800 psi. The automatic trip was caused due to the dip in RCS pressure below the Reactor Protective System setpoint. The Diamond Power Panel, the ICS Reactor Master, Steam Generator Reactor Master, both Feedwater (FDW) Loop Masters, and the delta Tc Station were in manual.

Post-trip response was mostly as expected. Pressurizer level decreased to 73 inches immediately after the trip, then increased to 285 inches. Operations personnel took action to lower the level to 160 inches. RCS pressure dropped to a low of approximately 1800 psig then increased and was controlled at 2175 psig. Hot and cold leg RCS temperatures converged and stabilized at approximately 555 degrees Fahrenheit, and no appreciable change was seen in RCS flow. Due to a fluctuation in feedwater flow, RCS Tave decreased momentarily to 540 degrees but was later stabilized after feedwater problems were corrected.

Both Steam Generator (S/G) levels initially dropped together to 150 inches and, at 8:49:53 hours, S/G "A" level continued to drop to about 25 inches and S/G "B" began to rise. RO "B", upon noticing a possible overfeed condition on the "B" Steam Generator, attempted to take manual control of the Main Feedwater Control Valve (FDW-41) and the Startup Feedwater Control Valve (FDW-44) in an attempt to decrease feedwater flow. However, a problem existed with FDW-41 and only after several attempts to manually control it, did the operator get FDW-41 in manual. Manual control of FDW-44 was obtained and the valve was closed. When FDW-44 was 50% closed, the Main Feedwater Block Valve, FDW-40, closed, terminating the overfeed condition.

A failure in the "B" Feedwater Controller (FDW-41) resulted in an overfeed of the "B" Steam Generator to about 75% on the operating range (see attachment # 3). This failure was attributed to the integral module controlling FDW-41, which was found set at 0.3 repeats/minute. The expected setting is 4.5 repeats/minute. The automatic demand signal sent to FDW-41 initially closed the valve. However, the large error across the integral (due to the incorrect setting) caused the valve to re-open. Then, upon operator input to close the valve, a much slower than normal response resulted. The valve response was verified to be consistent with the integral's as-found rate setting subsequent to this event.

During the time that FDW-41 was cycling back open, attempts to take FDW-41 to manual were delayed due to the installation of an improper relay module in the Integrated Control System (ICS). A type "G" relay module was used where a type "F" module was required. Upon inspection it was noted that the two modules were identical except that the "G" module had a time delay of 200 milliseconds to dropout. This also resulted in a longer time for the Hand/Auto station to respond to operator input. At approximately ten percent open FDW-41 was taken in hand and closed. It was later verified that no other situations of this nature existed in the ICS.

Also, the "A" Feedwater Pump Turbine (FWPT) tripped on high discharge pressure. The post-trip investigation found that the setpoint adjustment feature was loose, allowing the trip setpoint to drift from its normal setpoint of 1275 psig to 1175 psig. The discharge pressure setpoint was consequently calibrated to 1275 psig, and then fixed in place with a lock-tight adhesive (Work Request 25961C). The "A" FWPT high discharge pressure trip setpoint is normally set at 35 psi higher than that of the "B" FWPT. However, due to the setpoint drift, the "A" FWPT tripped and the "B" FWPT did not. No emergency feedwater actuation was required. The type of pressure switch [EIIS:XIS] used in this application has been noted in the past for its sensitivity and tendency to drift. Further investigation is planned to determine any generic implications of drift in this type switch since it is used throughout the secondary system.

Unit 3 was stabilized at hot shutdown conditions with Operations personnel safely controlling the reactor after the trip. No Engineered Safeguards Systems [EIIS:JE] or pressurizer relief valve actuation occurred, and no noticeable increases in RCS leakage were introduced. The transient is classified under Allowable Operating Transient Cycle type 8B.

CONCLUSION

The control rod power supply test, conducted on Unit 3, is a routine monthly test conducted on all critical Units at Oconee. Instrument and Electrical (I&E) Technicians "A" and "B" and Reactor Operators (RO) "A" and "B" were familiar with the procedure and qualified to perform the task. Work Request 55741A was issued in an attempt to detect and correct any electrical problems prior to the required rod movement test.

Immediately following the reactor trip, a station investigating party was formed to assess the causes and effects of the trip. The following information was reviewed during this LER investigation and is hereby included:

During the course of the immediate reactor trip investigation by plant personnel, attempts to re-create the failure of the normal (regulating) power supply to disconnect from the auxiliary power supply were unsuccessful. However, four possible failure sources were identified during the investigation:

(A) Failure of RO "B" to press the Clamp Release pushbutton on the Diamond to release the normal power supply.

(B) Failure of the 4-pole pushbutton switch to release the normal power supply even though the proper pushbutton was pressed and the light indicator on the Diamond was lit.

(C) Failure of the circuitry relay to actuate properly.

(D) Failure of the contactors to un-clamp properly.

According to the investigation findings, failure (A) and (D) were considered highly unlikely since the operator was observed to depress the clamp release button and the contactors involved are redundant on the regulating banks. The conclusion was reached that a spurious failure in the circuit could have occurred. This conclusion was reached by the investigating party after several hours of troubleshooting efforts, and if correct, indicates that a momentary equipment malfunction occurred.

During this LER investigation, it was determined that the most probable single failure is that the K7 Clamping Contactor momentarily failed. Failure of this contactor to open during this event would have prevented CLAMP RELEASE (disconnect from the auxiliary power supply). However, due to the working mechanism of the clamping contactors, it is extremely unlikely that one could fail to de-energize and unclamp once the power is removed and upon subsequent testing, not repeat the failure. The next most likely failure is double failure of both the K120 (CLAMP RELEASE) Contactor and the K108 (GROUP/AUXILIARY) Contactor. This possibility, while conceivable, is considered highly unlikely due to the fact that simultaneous double failures are extremely rare. Also the possibility that both could fail during the test and upon subsequent testing neither of them failed is very difficult to fathom.

To prevent a recurrence of this event, station management initiated a revision to the I&E procedure requiring the technician to stop any time three or more phase indicators lamps (located in CRD regulating supply cabinets in the cable room) are lit for longer than it takes the normal phase-to-phase transfer to occur. This revision will also require the I&E technician to physically travel to the control room for verification of the CLAMP RELEASE light. In addition, a double sign-off of the verification step will be required by the I&E technician and the operator.

Further, management will contact other utilities and/or other Babcock and Wilcox plants to gain any available information on similar events.

In conducting this LER investigation, an additional test was requested and performed on the Oconee simulator to see what would happen on the Diamond Power Panel when the procedure steps required in OP/0/A/1105/09, enclosure 4.2 (Operations procedure), were performed with the exception of step 8 under section 2.1. Step 8 states "Press selector for CLAMP RELEASE pushbutton." This is the same procedure that RO "B" used to set-up for the test on January 19, 1990. The results of this practical test revealed that even if step 8 was inadvertently skipped, that upon performing the next step, step 9 (Press selector for GROUP), that the CLAMP RELEASE light on the Diamond would come on. This finding was discussed with I&E Engineer "A" to ascertain whether or not a CLAMP RELEASE light would in all cases mean a CLAMP RELEASE had taken place (assuming no equipment failures occurred). I&E Engineer "A", after review of various electrical drawings, confirmed that a light on the Diamond Control Panel would mean CLAMP RELEASE had occurred (the normal power supply and the auxiliary power supply were disconnected). This evidence (even though based from a test conducted on the Oconee Simulator) indicates that the Diamond Power Panel in the situation used on January 19, during this event, was for the most part foolproof against possible operator errors of not performing certain required procedure steps. However, it is not foolproof against not performing all of the steps. According to I&E Engineer "A" the design is to ensure that a command signal from the Diamond Panel to move a group while two supplies are clamped will in fact unclamp the two supplies. Failure to unclamp two supplies and then to provide a command signal from within one of the supply cabinets in not inherently guarded against in the design.

Due to the possibility of the same results on Unit 3 that were obtained on the simulator, it is felt only reasonable to preclude the possibility of an inappropriate action on the part of RO "B". This decision is further supported by the fact that during the post-trip investigation, it was found that the operators were observed to depress the clamp release button. RO "B" stated, during this LER investigation, that he did press the CLAMP RELEASE pushbutton at the required time and RO "A" stated that the light was on when he verified the clamp release to I&E Technician "A". Also, during the post-trip investigation, the CLAMP RELEASE pushbutton was tested at least 100 times and no failures to unclamp were noted. Therefore, the root cause of this event is classified as UNKNOWN, Possible Equipment Malfunction.

Also, after the initial malfunction (power supplies not being disconnected), additional actions were needed to cause the reactor trip. Specifically, it was necessary for the normal power supply to be stepped through its phases as directed by IP/3/B/0340/02. During performance of these steps, the condition of four lights being lit simultaneously was observed and recognized as a condition not previously observed and not anticipated or associated with any known failure mechanism. The procedure in use gave no guidance for appropriate action in the event that more than three lights are lit at one time. The technicians recognized this and decided that the best course of action was to continue to sequence the power supply until it was returned to its original state (aligned with the auxiliary power supply), prior to troubleshooting. This action ultimately led to the reactor trip. The technician's decision was based on their understanding that the control rods were fully transferred to the auxiliary power supply and that any work they performed in the cabinet could not actually effect the control rods. A decision which could have avoided the reactor trip would have been the recognition that conditions which are unanticipated and not fully understood are cause to question all associated conditions, even those which are felt to be understood. This may have led to an investigation of the power supply separation prior to stepping the power supply back to its original condition. It is I&E management's opinion that there is an equal chance that any qualified technician would have made the same decision made in this case. A contributing cause of Management Deficiency, Deficient Policy is assigned. It is recommended that actions taken following observation of conditions which are not anticipated and not fully understood be discussed with all I&E technicians concerning management's expectations in such cases.

It was discovered in the initial investigation that the proportional plus integral module controlling FDW-41, found set at 0.3 repeats/minute, was reset during the recent end-of-core-11 refueling outage. The calibration and/or surveillance procedures associated with these modules are to be reviewed for possible revisions to include as-left calibration data. Documentation of changes to standard settings for the purposes of

Integrated Control System tuning will also be evaluated. I&E procedure, IP/0/B/0325/02, "ICS FW Control, B Loop Main FW and Startup FW Valve Calibration", will be changed to require a sign-off verification of the repeats/minute setting. It could not be determined during this investigation how the 0.3 repeats/minute setting was established, only that the last calibration was performed during the refueling outage in December 1989. A review of timed data available on the calibration data work sheet indicated that the setting was made properly. The cabinet housing this component is maintained locked with keys controlled by both the I&E and Operations Sections.

Additionally, the fact that an incorrect relay module was in use, which resulted in FDW-41 not properly responding to operator attempts to take the valve to manual, caused station restrictions to be placed on the issue of these modules. Specific approval was required for their use until an investigation was completed to determine how and why type "G" modules were substituted for type "Fs". Station I&E personnel later modified six type "G" modules, converting them to a type "F", and dispensed with the remaining "on hand " inventory of type "G" modules. The I&E procedure, to replace this module, provides adequate data to properly identify the correct module. However no requirement is stated in the procedure to compare or otherwise ensure exact replacements. Also in this case, the module was stamped appropriately, making it easy to identify. The installation date of the type "G" module was not found, but it is believed to have occurred several years ago. I&E management has agreed that I&E technicians should ensure correct replacements are made. Corrective actions to improve upon this situation in the future will be taken. The warehousing of incorrect relay modules was also investigated. The results indicated that a part number change, probably resulting from a manufacturer's module revision, led to an error in the next purchase requisition for the relay modules. This mix-up has apparently existed for sometime and went undetected due to the infrequent use of the modules.

The type of pressure switch used on the "A" Feedwater Pump Turbine to allow and establish a high discharge pressure trip setpoint is also under further investigation to determine any generic implications of setpoint drift. These switches are currently used throughout the secondary systems. However, it is possible that some may be replaced in conjunction with a subsequent modification.

The failure of the high pressure trip switch associated with the 3A FWPT is NPRDS reportable. The switch is manufactured by Meletron Company under manufacture model number 312-6SS-49A.

A review of station events occurring during the last 12 months revealed 4 other similar reactor trips; 2 involving Management Deficiency and 2 for

Unknown causes. These events were assigned the following LER numbers: 270/89-02, 270/89-03, 270/89-04 and 287/89-02. None of these events were associated with control rod group or single rod failures, therefore the corrective actions could not have prevented the possible equipment failures suspected in this event. The additional fact that a contributing cause of Management Deficiency, as identified in this incident, is also different from the other events, in that, the second cause was deficient policy. This second cause (deficient policy) was not determined in any of the previous reports. Therefore, this event is classified non-recurring.

The corrective actions subsequently taken and those that are planned should decrease the probability of a similar event in the future. There were no radioactive releases, personnel injuries or radiation exposures associated with this event.

CORRECTIVE ACTIONS

Immediate:

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

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

Subsequent:

1. Various Work Requests were prepared to correct specific post-trip problems.

2. A post-trip review and transient analysis were performed resulting in re-start of the Unit on 1/20/90 at 0145.

3. Instrument and Electrical (I&E) procedure (IP/3/B/0340/02) has been revised to provide the user with additional information. New steps were incorporated to include a double verification of CLAMP RELEASE on the Diamond Power Panel.

4. I&E procedure (IP/0/B/0325/02) was changed to require a sign-off verifying the repeats/minute setting is made correctly on the integral and the as-left calibration data is recorded.

Planned:

1. Production Support Training Department will develop training for operators that will test their response on the simulator to a scenario designed to duplicate the dropping of a rod group, reactivity effects, and Feedwater run-back failures.

2. The Operations Support Manager and his staff will evaluate the need for an audible alarm in the control room that will provide an additional, more aggressive indication of control rod group failure.

3. I&E management will provide additional guidance to all I&E technicians on requirements to stop work when at any time they become unsure as to the expected outcome of the procedure or steps within the procedure. The method for conducting this corrective action will be determined by the I&E Section Manager.

4. I&E management will communicate to all I&E technicians the importance of ensuring correct replacement parts are used. The method for conducting this corrective action will be determined by the I&E Section Manager.

5. Maintenance Engineering Section Manager will ensure that the Meletron Switch used on the 3A Feedwater Pump Turbine is investigated to determine any generic implications of drift. Replacement/modification of the switch will be based on the investigation results.

6. Maintenance Engineering Section Manager will ensure any available information related to CLAMP RELEASE failures at other utilities (especially other B&W units) is obtained and used appropriately.

7. Station Directive 2.2.1, section 9.0, "Procedure Use and Adherence", step 9.5, will be revised by Station management to clearly define their full expectations to implement this policy.

8. Training will be provided to all appropriate personnel on the revised Station Directive 2.2.1 (see corrective action #7).

SAFETY ANALYSIS

Unit 3 tripped from 49% Full Power (FP) on January 19,1990 following an immediate reduction in power from 100% FP. This happened as a result of Control Rod Group 6 dropping into the core. Initially the unit ranback to 38% FP followed by a 10-15% power increase. The unit automatically tripped on low Reactor Coolant System (RCS) pressure (approximately 1800 psig) approximately 18 seconds after the rod group dropped. The Integrated Control System had earlier been placed in manual for the control rod power supply test. Post-trip Feedwater (FDW) control problems initially prevented the operators from manually running back the FDW demand resulting in an overfeed of the "B" Steam Generator to approximately 75% on the operating range.

Only the dropping of one control rod is analyzed in the Final Safety Analysis Report (FSAR), section 15.7, "Control Rod Misalignment Accidents". The Duke Power, Design Engineering, Nuclear Engineering Support Section generally feels that the dropping of a group of rods, while not analyzed, would make it very difficult for the reactor to successfully runback to a lower power level and not trip. The manual or automatic trip of the reactor terminates the initial transient and prevents the reactor from exceeding monitored parameters. Station Operation procedures require the manual trip of the reactor if more than one control rod drops. Operators were in the process of manually tripping the Unit on January 19 when the Reactor Protective System automatic trip was initiated. Reactor tilt/imbalance related problems (caused by a group drop) are less significant than the consequences of a single rod drop. This is due to the distribution of the group rods in the core.

The consequences of the "B" Steam Generator overfeed could lead to overcooling of the reactor coolant system. The first line of defense is Operator action. Given this situation, the operator would take manual control of the Main Feedwater Control Valve, FDW-41, and attempt to control the level of the steam generator. In accordance with the Emergency Operating Procedure, if overcooling continues due to rising steam generator level, the Operator is directed to trip the Main Feedwater Pumps and control steam generator level with the Emergency Feedwater Pumps (which were operable during this event). The next level of defense is that the Main Feedwater Pump will automatically trip on a high steam generator level. Should the automatic trip fail, the Operations Management Procedure directs the operator to trip the Main Feedwater Pump at 96% on the operating range for the steam generator. During this event, FDW-44 (Startup Feedwater Control Valve) was placed in manual and then closed. When FDW-40 was 50% closed, the main block valve (FDW-40) automatically closed terminating the overfeed condition.

NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	9 0	— 0 0 1	— 0 0	1 3 OF 1 6

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

The health and safety of the public was not jeopardized due to this incident and the operators safely controlled the trip and responded to the feedwater abnormalities in a satisfactory manner. The Unit was placed in hot shutdown conditions and an investigation was performed to assess the cause and effects of the trip. The Unit was restored to power on January 20, 1990.

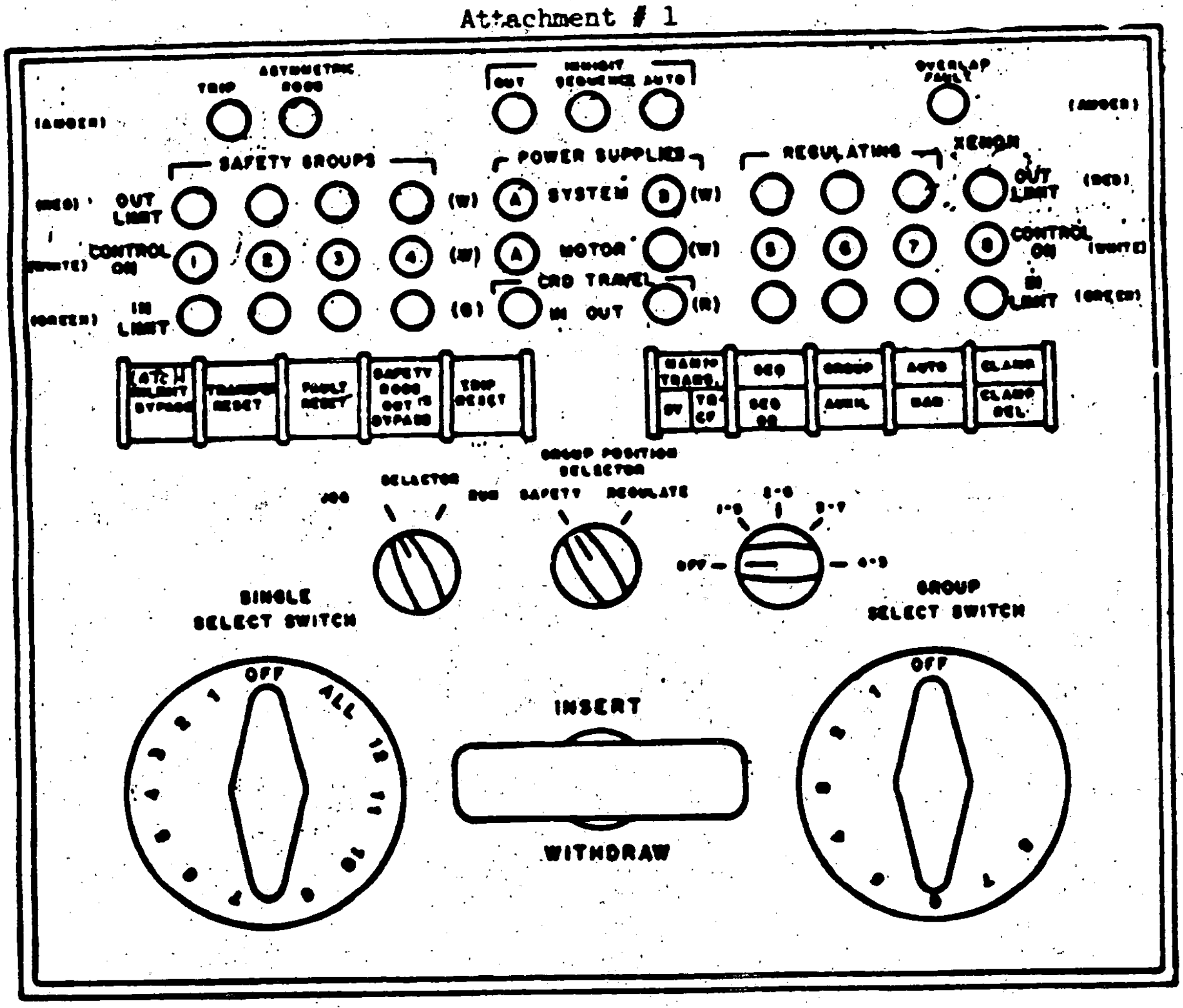
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	9 0	0 0 1	0 0	1 4 OF 1 6

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

Attachment # 1

ATTACHMENT #1



Control Rod Drive Instrumentation	Operator's Control Panel (Diamond)	OC-IC-CRI-9 2-20-85 B&W Training Manual TLF/ARB

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit 3	0 5 0 0 0 2 8 7	9 0	— 0 0 1 —	0 0	1 5 OF 1 6

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

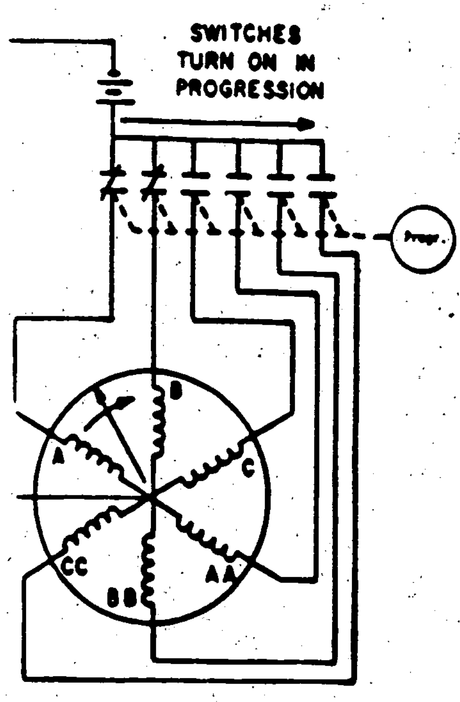
ATTACHMENT # 2

SUMMARY DESCRIPTION AND ILLUSTRATION OF TEST SEQUENCE CAUSING ROD GROUP 6 TO DROP INTO THE CORE

Summary Description:

During testing of the Group 6 control rod power supply, transfer of power to the auxiliary power supply was initiated. The normal power supply apparently failed to disengage properly. As a result of this failure, when the normal power supply was cycled to the various phases of the rod contactors (as illustrated below), four phases were energized instead of the normal two phases. Normal operation is that the next phase in sequence is energized and therefore three phases are energized for a brief time until the first phase in the sequence drops out due to the three-two hold feature. At the beginning of the test phases, AA and BB were energized. After cycling once, phases AA, BB, and CC were energized. After cycling again, phases AA, BB, CC, and A were energized (phase AA would have de-energized if the normal power supply had been disconnected). Cycling one more time, phases AA, BB, A, and B were energized. When this occurred, the magnetic fields were 180 degrees out of phase effectively cancelling each other, and the lead screw roller nuts disengaged resulting in the rods in group 6 falling into the core.

Illustration:



NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

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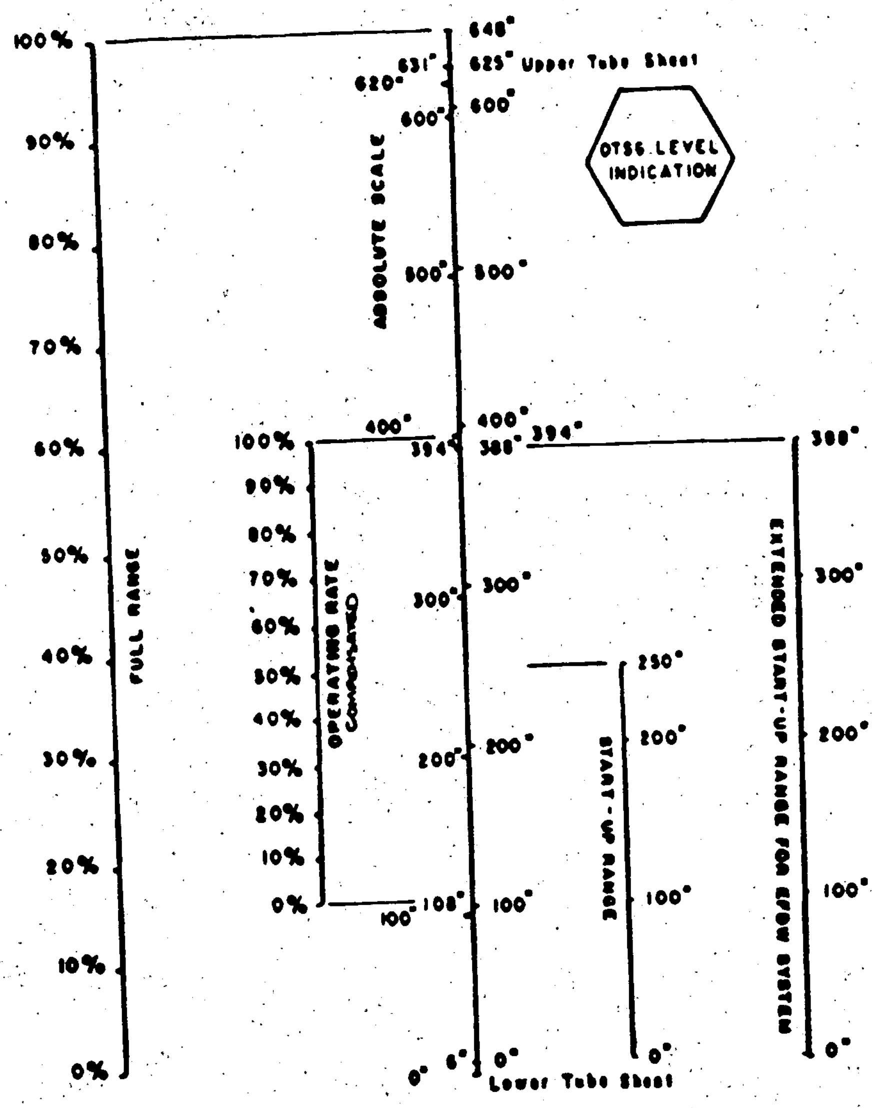
TEXT (If more space is required, use additional NRC Form 366A's) (17)

ATTACHMENT # 3



ONCE-THROUGH STEM GENERATOR	OTSG LEVEL RANGES	OC-CM-SG-12 / 2-13-85 / B&W TECH MANUAL / JPO/ARB / TRAINING USE ONLY