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FACIL: 50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270

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SUBJECT: LER 93-007-00: on 931024, Unit 2 tripped form 100% full power on flux/flow/inbalance trip of RPS. Caused by technical deficiency. Core operating limit calculations reanalyzed & calibr procedures revised. W/931123 ltr.

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November 23, 1993

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
LER 270/93-07

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 270/93-07, 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,

J W Hampton

J. W. Hampton
Vice President

/ftr

Attachment

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IE22 1/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 2	05000 270	1 OF 9

TITLE (4)

Inappropriate Action, Deficient Procedure, And Low Flow Spike Result In Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
10	24	93	93	-- 07 --	00	10	23	93	FACILITY NAME	DOCKET NUMBER 05000

		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
OPERATING MODE (9)	N	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	100	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 (Include Area Code)
S. G. Benesole, Safety Review Manager	(803) 885-3518

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 24, 1993, at 0634 hours, Unit 2 tripped from 100 % full power on a flux/flow/imbalance trip of the Reactor Protective System (RPS). The flux/flow/imbalance trip occurred when transmitters monitoring flow in both loops of the Reactor Coolant System (RCS) simultaneously indicated that flow spiked low. Post trip operator response stabilized the unit. An investigation revealed that the method used to calibrate the RCS flow inputs to the RPS flux/flow/imbalance trip introduced an error which reduced the operating margin associated with the trip function. Also, it was found that an error was made while performing the last completed calibration procedure, which further reduced the margin. Therefore, the root causes of this event are considered to be Deficient Procedure (technical deficiency) and an Inappropriate Action (failure to follow procedure, improperly followed the correct procedure) in conjunction with an indicated low RCS flow spike. Corrective actions included reanalyzing core operating limit calculations to increase the operating margin, revising the calibration procedure, and reviewing this event with personnel involved in the procedure error.

BACKGROUND

The Reactor Coolant System (RCS) [EIIS:AB] has two steam generators with associated pumps, piping, and instrumentation. These are designated Loop A and Loop B. The flow indications for each loop are provided by a pair of impulse lines which run through a secondary shield wall inside the Reactor Building [EIIS:NH]. Each pair of impulse lines are connected to five differential pressure transmitters. One transmitter (Channel E) provides the RCS flow input signal to the Integrated Control System [EIIS:JA] and the Transient Monitor. The other four transmitters of each loop are connected to the four redundant channels of the Reactor Protection System (RPS) [EIIS:JC], designated as Channels A, B, C, and D. Each channel receives a total flow input which is the result of the sum of flow in each loop.

The RPS is a safety related system which protects the reactor core from potential damage by automatically deenergizing control rod drive mechanisms when two of four independent input channels reach their trip setpoint. One of the RPS trip parameters is Flux/Flow/Imbalance. The RPS compares the indicated neutron flux (i.e. power level), RCS flow rate, and the power imbalance (the power produced in the top half of the core minus the power produced in the bottom half of the core). The result is that there is a minimum RCS flow for any given power level, and the reactor will be tripped if the flow is less than that minimum on any two of the four RPS flow channels.

The Duke Power Company Core Operating Limits Report (COLR) establishes protected limits for reactor power to RCS flow. These limits also take into account power imbalance. The COLR, further provides trip setpoints relative to the flux to flow ratio. The flux to flow ratio trip setpoint was established at 107 %. Therefore, if the 107 % setpoint was used, a reactor trip would occur, with 100 % RCS flow and a reactor power increase to 107 % or with reactor power at 100 % and an RCS flow decrease to 93.46 % of full flow. These COLR setpoints are reduced to 105.5 % for implementation by procedure, to account for instrument drift.

The RCS flow input to the RPS flux/flow/imbalance trip is derived by converting the differential pressure (dp) output of each flow transmitter to a flow signal by processing the dp signal through a square root extractor. The individual loop flow signals are then summed to get a total RCS flow for each channel. This flow signal is then multiplied by a calibration constant (gain) to determine a trip setpoint equivalent to 105.5 % power with a flow equal to what is measured as 100 % flow.

The gain is verified on every reactor startup after refueling using procedure IP/0/A/305/04 (Reactor Protective System Flow Check). This procedure records the output of each square root extractor from all four RPS channels on each loop. The signature of the output of each square root extractor has signal noise which is generally within a band with random spikes outside of this band. Therefore, to obtain an average output an instrument is used which records the maximum and minimum output values. These maximum and minimum values from each transmitter are averaged to obtain RCS total average flow. This total flow value is then used to calculate the gain necessary to normalize this total flow (100 %) for a trip setpoint equal to a power level input of 105.5 % to the flux/flow/imbalance trip.

This procedure is performed twice on each startup after refueling, once at 75 % and once at 100 % power. Since the flow at 75 % power is 1 % lower than the flow at 100 % power, the average flow measured at 75 % power is adjusted up 1 % before the gain is calculated. This results in a more accurate prediction of full power values.

EVENT DESCRIPTION

On October 24, 1993 at 0534 hours, with Unit 2 at approximately 100 % Full Power, Reactor Protective System (RPS) Channel C tripped on flux/flow/imbalance. At 0535 hours, Channel C was reset after verifying that all parameters were normal. At this time, the Control Room Operator noticed that the Nuclear Instrumentation (NI) indicated approximately .5 % greater than actual thermal power, which is within allowable limits but made plans to have the NIs recalibrated later that day.

At 0634:09 hours Unit 2 tripped from approximately 100.5 % full power as indicated by NIs. The reactor trip occurred when all four channels of the Reactor Protective System (RPS) actuated on flux/flow/imbalance. All four channels tripped within 79 milliseconds as indicated on the events recorder. The Control Rod Drive breakers opened, and all full length control rods dropped into the core, shutting down the reactor. The turbine/generator tripped, station auxiliaries [EIIS:EA] switched from normal to start-up power, and the Main Steam Relief Valves opened. All four Turbine Bypass Valves (TBV) opened, however one of the four valves was previously isolated due to seat leakage.

The operators confirmed that the Reactor and Turbine had tripped and monitored for proper operation of other automatic equipment. The Operators entered the Emergency Operating Procedures, and as normally required after a reactor trip, manually started High Pressure Injection (HPI) [EIIS:BG] pump "A" at 0634:46 hours and opened 2HP-26 (HPI Loop A Emergency Make-up

Valve) to increase HPI flow to maintain Pressurizer level. At 0636:56 hours, the operator closed 2HP-26 and stopped HPI pump "B".

Post trip plant response was generally as expected. Following the reactor trip the average Reactor Coolant System (RCS) temperature decreased from 579 F and stabilized at 552 F. RCS pressure decreased from approximately 2145 psig to 1818 psig and then slowly increased to 2203 psig. Pressurizer level decreased to a minimum of 63 inches then increased and stabilized at approximately 138 inches. Steam Generator (SG) "A" pressure reached a post-trip high of 1138 psig and a minimum of 995 psig before stabilizing at 1012 psig. SG "B" pressure reached a post-trip high of 1130 psig and a minimum of 983 psig before stabilizing at 1012 psig. SG levels decreased to a minimum of 22 inches then were increased and maintained at 25 inches by main feedwater.

The power was lost to the Radiation Monitor (RM) control room indications momentarily due to the affect of transferring auxiliary power sources on the monitoring computer. This has been previously identified as a problem on all three units. After the transfer, the computer reset itself and indication was restored within a few minutes. This problem solely affected the control room indication. The RM was available throughout this event. The data was accessible at the processor skid.

The transient monitor data was reviewed and indicated a low RCS flow spike occurred just prior to the reactor trip. The lowest measurable spike recorded (by the transient monitor which measures at 300 millisecond intervals) at the time of the trip was found to be an indicated 4.2 % reduction in flow. It was concluded that the transient monitor system had not actually seen the true "bottom of spike" due to the relatively large sampling frequency (compared to the width of the observed spike). The following actions were performed to determine whether the indicated flow spike was due to a real flow reduction.

The RCS flow transmitters and impulse lines were inspected for leaks, and none were found.

Since RPS Channels B and C were the first channels to trip, the instrument strings were checked while shutdown and found to be operating properly.

An evaluation of Unit 2 RCS flow rates available from trending data since the beginning of cycle were evaluated and it was concluded that flow was consistent and non-changing throughout the cycle. No abnormalities were identified with long term flow trends.

Other plant systems/parameters (i.e. system voltage, Reactor Coolant Pump parameters, 125 VDC system and Loose Parts Monitoring System) were reviewed for abnormalities which would cause or indicate a flow reduction and none were found.

Based on the investigation it was determined that the simultaneous occurrence of an indicated low flow spike in RCS loops A and B, concurrent with NI power high in the control band, caused the trip. It was decided, based on the information above, that a Reactor restart and power escalation to 98 % could be conducted without undue risk of an additional transient or challenge to fuel integrity, and that further investigation could continue at that reduced power level. This decision was based on the evaluation that RPS RCS flow rates were verified to be normal for the existing shutdown condition and there was no basis to assume power operations would be different from normal.

The decision to restrict power to 98 % FP was based on assuring an adequate margin to trip and the desire to obtain further flow data at a condition "as close as possible" to the condition at which the trip occurred. It was determined that 98 % FP would provide a conservatively large flux/flow margin adequate to prevent another reactor trip, with a recurrence of the flow conditions which existed at the time of the trip.

A qualitative evaluation was conducted as to why NI power had slowly ramped up from approximately 0400 hours that morning until the time of the trip (from approximately 99.9 % to approximately 100.5 %). It was concluded, based on the evaluation, that this was a result of normal Integrated Control System (ICS) control band characteristics.

On October 25, 1993, at 2340 hours, Unit 2 was returned to 98 % full power and RCS flow inputs to the RPS flux/flow/imbalance trip were measured. The comparison of the Unit 2's trip setpoints to the Unit 1 and 3's setpoints revealed that Unit 2 had the lowest setpoints. The Unit 1 setpoints were in the normal range and the Unit 3 setpoints were low.

Further investigation revealed that a procedural inadequacy existed which caused the low RCS flow inputs on Unit 2 and 3. The method of measuring the average flow output of the square root extractors did not ensure a true average. Using the average of the maximum and minimum values resulted in a shift in the calculated average flow value when the peak values were biased by a high or low spike. This was evident in the Unit 2 and 3 "as found" data.

Also, it was discovered that an error was made while performing IP/0/A/305/04 (Reactor Protective System Flow Check) on July 13, 1993. This error resulted in Unit 2's average trip setpoints being the lowest.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

While performing IP/0/A/305/04 on July 13, 1993, with Unit 2 at 100 % power, Instrument and Electrical Technician A used the enclosure in the procedure for 75 % power. This enclosure is used to adjust the 75 % flow data up to the expected 100 % flow values. Using this formula with 100 % data incorrectly shifted the data up resulting in a lower calibrated gain value. This lower gain resulted in a further reduction in the margin of flux to flow on Unit 2.

IP/0/A/305/04 was revised to change the method for obtaining the flow voltage data used in the gain calculation. Also, instructions were added to perform a post calibration verification. On October 27, 1993 Unit 2 setpoints were recalibrated to regain the margin to trip, then Unit 2 was returned to 100 % full power. Unit 3 was recalibrated on October 28, 1993.

CONCLUSIONS

The root causes of the trip of Oconee Unit 2 are considered to be a Deficient Procedure (technical deficiency) and Inappropriate Action (failure to follow procedure, improperly followed the correct procedure) and an indicated low Reactor Coolant System (RCS) flow spike. The procedure methodology and inappropriate action reduced the margin of flux to flow and the spike was significant enough to reach the trip setpoint with the reduced margin.

Plant data such as switchyard voltage, Reactor Coolant Pump (RCP) parameters, 125 VDC system and loose parts monitoring system was reviewed. This data did not indicate any perturbations that would cause a real reduction in RCS flow. The indicated decrease in RCS flow was approximately 4.2 % with a duration of less than 300 milliseconds. It is not believed that a real decrease in RCS flow of this magnitude and duration is physically possible. The RCP flywheels, which prevent any drastic changes in pump speed and the momentum of the RCS fluid make the possibility of a real RCS flow change like the one indicated highly unlikely. Had any structures within the reactor core shifted to block existing flow channels, some indications on the loose parts monitoring should have been noted. Finally, review of past RCS flow data revealed that process noise is to be expected within certain bands. However, spikes outside of these bands do occur. Other Babcock and Wilcox (B&W) plants were contacted to determine if the flow noise seen at Oconee was consistent with that seen at other plants. It was found that other B&W plant's flow noise was similar to that seen at Oconee. Based on the above, it is concluded that the Reactor trip was due to a spurious indicated low flow signal and not a real flow reduction.

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 2	05000 270	93	- 07 -	00	7 OF 9

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

The procedural inadequacy in the method used to calibrate the RCS flow input to the RPS flux/flow/imbalance trip module introduced an error. In obtaining the individual loop flow averages, peak to peak averages were used instead of time weighted averages. This method led to an unrepresentative 100 % flow value and therefore reduced the margin of flux to flow. Additionally, during the last performance of the calibration procedure an error was found which also reduced the margin of flux to flow. This error was due to an Inappropriate Action, failure to follow procedure.

The calibration procedure was changed to calculate a more accurate average and to add a post calibration verification.

A Problem Investigation Process (PIP) report (0-093-0375) was outstanding prior to this reactor trip to address the momentary loss of Radiation Monitor indications in the control room following a trip. As a result of this PIP, a minor modification was completed on Unit 2 on October 28, 1993 that should prevent the loss of Radiation Monitor indications in the control room following reactor trips.

An investigation into the cause of Main Steam (MS) pressure reaching 1138 psig following the reactor trip was initiated. The investigation has included a review of data of previous trips on all three units. This review has revealed that the higher than expected Steam Generator (SG) outlet pressure is not unique to this reactor trip. Previous trips on Units 1 and Unit 2, indicate SG outlet pressure or MS pressure have exceeded 1115 psig. Trips on Unit 3 have reached as high as 1111 psig in the last seven trips. The initial theory for the increase in SG Outlet Pressure on this trip was attributed to the isolation of one turbine bypass valve. After a Unit 2 reactor trip, on October 19, 1992, a SG Outlet pressure of 1124 psig was observed. This trip occurred prior to the isolation of the TBV, thus indicating that a TBV isolation was not the major contributor to the observed response of S/G outlet pressure. It should be noted that all these pressures are within the design basis limits. Historical setpoint drift of the Main Steam Relief Valves (MSRV) on Unit 2 was reviewed. All sixteen MSRVs are tested on startup after each refueling outage. A review of Unit 2's last test shows all sixteen MSRV's "As-found" lift pressures were less than the setpoint, however, four of the sixteen were rebuilt prior to the test as part of a scheduled preventive maintenance program. Thus, high setpoint drift was not a prior problem. The investigation into the response of S/G outlet pressures is continuing.

This event is not considered to be recurring. There have been no unit trips on flux/flow/imbalance due to a deficient procedure or inappropriate action within the last two years. However, there have been spurious Reactor Protective System flux/flow/imbalance channel trips, like the

Channel C trip approximately one hour before this reactor trip. As a rule these spurious channel trips have been investigated with the root cause of these channel trips having been determined to be equipment failures, but some were classified as unknown. Therefore, some of these unknown individual channel trips may have resulted from the procedural inadequacy identified during this event.

There were no NPRDS reportable equipment failures associated with this event. The event did not result in radioactive releases, overexposures of radiation, or personnel injuries.

<u>CORRECTIVE ACTIONS</u>

Immediate

1. Operators took appropriate actions to stabilize the unit at hot shutdown.

Subsequent

1. IP/0/A/305/04 "Reactor Protective System Flow Check" procedure was revised to enhance the flow averaging method and to perform a post calibration verification.

2. Instrument and Electrical personnel adjusted flow gain on Unit 2 and 3's Reactor Protective System channels so that the value was closer to the 100% full power flow value. Unit 1 did not require resetting.

3. Safety Analysis reanalyzed RCS flow calculations to increase the margin from flux to flow.

4. This event was reviewed with technicians involved in the procedure error.

Planned

1. Component Engineering will obtain software to perform A-D conversion and quantification of the Reactor Protective System/Reactor Coolant System flow channel output to enhance data collecting and analysis.

2. Engineering will quantify the level of noise generated by Reactor Coolant flow to verify a tolerance to be used in the calibration procedure.

3. Component Engineering will review this event as it applies to other Instrument and Electrical procedures that utilize process inputs for calibration and make required changes.

4. Instrument and Electrical will evaluate the Reactor Protective System Flow Check procedure to determine if enhancements can be made to reduce personnel errors and make the required procedure changes.

SAFETY ANALYSIS

This event was initiated by a trip on flux/flow/imbalance for which the setpoint was verified to be conservative. The Reactor Protective System operated as designed and tripped the unit. The plant post-trip response was normal except Steam Generator A and B post-trip pressure was considered slightly higher than normal, but Main Steam Relief valves adequately controlled pressure such that design pressure was not exceeded. No Engineered Safeguards System or Emergency Feedwater actuations were either required or received. The health and safety of the public was not compromised by this event.