ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9207170067 DOC.DATE: 92/07/09 NOTARIZED: NO DOCKET #
FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
AUTH.NAME AUTHOR AFFILIATION
BENESOLE,S.G. Duke Power Co.
HAMPTON,J.W. Duke Power Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 92-005-00:on 920508,unit tripped from 14% full power & emergency feedwater actuated.Caused by equipment failure & defective procedure.Solenoid valve replaced & post-trip review directive revised.W/920709 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-3 LA	1	1	PD2-3 PD	1	1
	WIENS,L	1	1			
INTERNAL:	ACNW	2	2	ACRS	2	2
	AEOD/DOA	1	1	AEOD/DSP/TPAB	1	1
	AEOD/ROAB/DSP	2	2	NRR/DET/EMEB 7E	1	1
	NRR/DLPQ/LHFB10	1	1	NRR/DLPQ/LPEB10	1	1
	NRR/DOEA/OEAB	1	1	NRR/DREP/PRPB11	2	2
	NRR/DST/SELB 8D	1	1	NRR/DST/SICB8H3	1	1
	NRR/DST/SPLB8D1	1	1	NRR/DST/SRXB 8E	1	1
	REG FILE 02	1	1	RES/DSIR/EIB	1	1
	RGN2 FILE 01	1	1			
EXTERNAL:	EG&G BRYCE,J.H	3	3	L ST LOBBY WARD	1	1
	NRC PDR	1	1	NSIC MURPHY,G.A	1	1
	NSIC POORE,W.	1	1	NUDOCS FULL TXT	1	1

R I D S / A D D S

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

R I D S / A D D S

FULL TEXT CONVERSION REQUIRED
TOTAL NUMBER OF COPIES REQUIRED: LTTR 32 ENCL 32

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

J.W. HAMPTON
Vice President
(803)885-3499 Office
(704)373-5222 FAX

DUKE POWER

July 9, 1992

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

Subject: Oconee Nuclear Site
Docket Nos. 50-269, -270, -287
LER 269/92-05

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 269/92-05, concerning a Technical Specification violation.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(i)(B). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

HR Barron for

J. W. Hampton
Vice President

/ftr

Attachment

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

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

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

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

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

IE22

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oconee Nuclear Station, Unit 1	0 5 0 0 0 2 6 9	1 OF 0 8

TITLE (4) Equipment Failure and Defective Procedure Result In Operation In Violation of Technical Specification

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 5	1 0	9 2	9 2	0 0 5	0 0	0 7	0 9	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): - 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)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)(B)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER AREA CODE	
S. G. Benesole, Safety Review Manager	8 0 3	8 8 5 - 3 5 1 8

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
F	B A	X C V	V 0 3 0	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [N] NO

EXPECTED SUBMISSION DATE (15) MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT

On May 8, 1992, at 0342 hours, Unit 1 tripped from 14% full power and Emergency Feedwater (EFDW) actuated. The Main Feedwater pump did not trip and the operators secured the Motor Driven EFDW pumps.

The Post-Trip Review, Reactor Transient Analysis, and the subsequent Licensee Event Report did not identify the fact that flow did not exist in the EFDW train A that contains control valve (1FDW-315). Technical Specifications require two flow paths to be operable when the unit is above 250 F. Seventeen days after the reactor was heated above 250 F the control valve (1FDW-315) was discovered to be inoperable, in the automatic mode, when a periodic stroke test was performed. Therefore, Unit 1 had operated outside of Technical Specification requirements.

There were two root causes for this event: Equipment Failure and Defective Procedure, Technical Deficiency. Corrective Actions include replacing the solenoid valve and revising the Post Trip Review Directive.

BACKGROUND

The purpose of the Emergency Feedwater (EFDW) [EIIS:BA] system is to remove decay heat and Reactor Coolant Pump heat following a loss of Main Feedwater (MFDW) [EIIS:SJ]. Three EFDW pumps are provided for each unit. Two motor driven pumps are powered by emergency AC power while the turbine driven pump is aligned to Main Steam [EIIS:SB] or Auxiliary Steam [EIIS:SA]. Each unit's EFDW system is designed to supply feedwater to the Steam Generators (SG) in the event MFDW is lost.

There are three systems at Oconee which are designed to automatically actuate when the setpoints of low MFDW pump hydraulic oil pressure and/or MFDW pump discharge header pressure are reached on both MFDW pumps. The systems are the EFDW system, the Reactor Protective System (RPS) [EIIS:JC] and the Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC). Each of these systems use diverse means to determine when MFDW has been lost. Each system actuates when signals are received that both MFDW pumps can no longer provide feedwater to the SGs. The EFDW system (all 3 pumps) will start automatically upon loss of both MFDW pumps (indicated by low MFDW pump turbine hydraulic control oil pressure of 75 psig and/or low MFDW pump discharge header pressure of 800 psig decreasing). This actuation will also enable a circuit which controls SG level [EIIS:JB] at predetermined setpoints (30 inches on the start-up range with Reactor Coolant Pumps in operation). The loss of MFDW provides a signal to the RPS as an anticipatory trip that trips the Reactor prior to Reactor Coolant System [EIIS:AB] parameters reaching their own trip setpoints. The pressure switches and/or AMSAC initiates the start of the EFDW pump turbine. If the start signal clears (i.e. MFDW pump discharge pressure increases above 800 psig) within 15 seconds +/- 1 second, the EFDW pump turbine will reset. The AMSAC signal will initiate the two Motor Driven EFDW pumps and trip the main turbine if it is on line.

EFDW control valve 1FDW-315 is a pneumatically-operated valve that regulates the flow of EFDW to SG A, for control of the water level. The 125VDC, three-way solenoid valve 1FDW SV0200 selects whether control of 1FDW-315 will be manual or automatic.

Technical Specification 3.4 requires two EFDW flow paths to be operable when the reactor is heated above 250 F. The flow path is defined in the Technical Specification Bases as: The flow path to either steam generator including associated valves and piping capable of being supplied by either the turbine driven or the associated motor driven pump. Additionally, the EFDW system is designed to start automatically upon receiving an initiating signal.

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

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

EVENT DESCRIPTION

On May 8, 1992, at 0145 hours, the Unit 1 Reactor was critical and preparations were being made to increase Reactor power and place the Electrical Generator [EIIS:EL] on line, following a previous Reactor trip (which was was reported in LER 269/92-03).

At 0325 hours, with the Reactor at 14% full power and the B Main Feedwater (MFDW) pump in service, problems were encountered with high Hotwell [EIIS:KA] level. While trying to lower the level in the Hotwell, the Reactor and Main Turbine [EIIS:TA] tripped at 0342:23 hours due to a feedwater transient. This event was reported in LER 269/92-04.

At 0342:23 hours, the A and B Motor Driven Emergency Feedwater Pumps (MDEFDWPs) started on a low Main Feedwater Pump (MFDWP) discharge pressure. The B MFDWP did not trip and Control Room Operator A (CRO-A) secured the MDEFDWPs at 0343:06 hours, after verifying proper operation of the B MFDWP. CRO-A stated that he also verified the Steam Generator levels were being controlled by the B MFDWP. CRO-A did not observe or verify flow through the two trains of Emergency Feedwater (EFDW). The Emergency Operating Procedure does not require the CRO to verify EFDW flow unless there is a loss of MFDW.

A Post-Trip Review Report was completed on May 8, 1992 by Shift Manager (SM) A, with assistance from SM-B, the Engineering Supervisor and the duty Reactor Engineer. SM-A noted in the Plant Response section of the report that the MFDWP trip signal had not occurred but the MDEFDWP A and B had started. The MDEFDWP start signal was from low MFDWP discharge pressure. The start and stop times for each pump were recorded. The Turbine Driven Emergency Feedwater Pump (TDEFWP) initiated, but the MFDWP discharge pressure went above the setpoint before the 15 second seal-in timed out. This satisfied the logic for the TDEFDWP. SM-A stated that, during his review, the failure of the TDEFDWP to start was questioned and verified to be the correct response.

On May 10, 1992, at 1509 hours, the Reactor Coolant System (RCS) temperature was increased to 325 F.

The Reactor was critical at 1517 hours on May 11, 1992. On May 12, 1992 at 1827 hours the Unit reached 100% Full Power. The Unit continued to operate at 100% Full Power until May 24, 1992, at 2010 hours, when a Reactor power reduction was begun to repair the 1A2 Reactor Coolant Pump Seals. The Reactor was shutdown at 0438 hours, on May 25, 1992. The RCS was cooled to < 250 F by 2040 hours.

On May 27, 1992, Performance Technicians performed the 1FDW-315 and 1FDW-316 Stroke Test procedure (PT/1/A/0150/22M). The test is performed on a "Quarterly at Cold Shutdown" frequency to determine operability of the automatic function of 1FDW-315 (Steam Generator A EFDW Control Valve) and 1FDW-316 (Steam Generator B EFDW Control Valve).

The 1FDW-316 valve stroke times were in the acceptable range but the 1FDW-315 valve failed to operate. A work request was issued to investigate and repair 1FDW-315 valve. Investigations by the Instrument and Electrical (IAE) Technicians revealed that the solenoid valve used for enabling the automatic functioning of 1FDW-315 had failed. The failures are due to the valve being energized continually, resulting in overheating and binding. This causes the control valve (FDW-315) to be inoperable in the automatic mode. This was also identified in LER 287/91-07.

On May 30, 1992, Station Management discussed the need and intent to review the Post Trip data with respect to 1FDW-315.

On June 1, 1992 the solenoid valve (1SV-200) was replaced with a newer model valve as directed by the previous commitment (LER 287/91-07) due to failure of the original solenoid valve.

On June 2, 1992, the stroke test (PT/1/A/0150/22M) was performed on 1FDW-315 valve after the solenoid valve (1SV-200) had been replaced. The valve operated and stroke times were in the acceptable range.

The IAE Section issued a Problem Report on June 4, 1992 for identification of 1FDW-315 not working in automatic. This was to document the fact that this was a repetitive failure.

The Reactor Trip (LER 269/92-04), reporting the May 8, 1992 Reactor trip, was approved and sent to the Nuclear Regulatory Commission on June 8, 1992.

On June 11, 1992, the Safety Review Section held discussions and reviewed data with the Reactor Engineering Group concerning the EFDW actuation, following the Unit trip of May 8, 1992. It was noted that the A EFDW train had exhibited no flow. From a more detailed review of existing Transient Monitor information, it was determined that 1FDW-315 valve had not opened.

The last time the 1FDW-315 valve stroke test was performed satisfactorily was September 22,1991, during a refueling shutdown.

CONCLUSIONS

There were two root causes associated with this event: Equipment Failure and Defective Procedure, Incomplete Information. Technical Specification 3.4.1.b requires two flow paths to be operable when the reactor is heated above 250 F. The reactor operated at power for 15 days with one flow path inoperable in the automatic mode.

The root cause of Equipment Failure was due to the failure of 1SV-200. This is similar to the event documented in LER 287/91-07. The valves in Unit 2 have been replaced. The valves in Units 1 and 3 are scheduled to be replaced during the next refueling outages. The solenoid valve failure is NPRDS reportable. The valve is a Valcor V-70900-21-3, Serial Number 1495. This root cause is considered recurring.

The root cause of the failure to identify that one Emergency Feedwater (EFDW) flow path was inoperable, is Defective Procedure, Technical Deficiency. If the Post Trip Review had explicitly required the verification of EFDW flow in each train this event could have been prevented. The fact that no flow was present in the SG A EFDW train could have been verified by a more detailed review of the transient monitor charts. Since this was not observed in the Post Trip Review, the approval to restart was made and the Unit was heated above the temperature that EFDW is required to be operable.

The safety systems which respond to a loss of Main Feedwater (MFDW) receive automatic actuation from the presence of a low MFDW pump discharge header pressure (800 psig) or low MFDW pump hydraulic oil pressure (75 psig) signal. The A MFDW pump was off and the B MFDW pump was supplying the feedwater to the Steam Generators (SG) at the time of the event.

The transient monitor plot (See Attachment A) for Emergency Feedwater (EFDW) flow that was submitted as part of the Reactor Transient Analysis was plotted on a 15 minute time line. The MDEFDWPs were on for approximately 43 seconds. The amount of time the EFDW controls called for 1FDW-315 and 1FDW-316 to be open was only 15-30 seconds. Unless the flow parameter had been observed by the Control Room Operator during this time frame, it would not have been detected. The personnel performing Post Trip Review and Transient Analysis stated that they did not place sufficient emphasis on the EFDW flow aspect since the B MFDW pump remained on during the event and both Motor Driven EFDW pumps started. They also observed that steam generator levels tracked together to approximately 18 inches immediately following the reactor trip and progressed to the level where EFDW maintains (30 inches).

The Post Trip Review Checklist did not explicitly require documentation that flow had been established in both SG trains. The transient monitor plot showing EFDW train A and B flow was not clear in showing that both EFDW trains had exhibited flow.

The Defective Procedure is considered recurring based on a review of Problem Investigation Report Database.

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

CORRECTIVE ACTIONS

Immediate

1. The solenoid valve (1SV-200) was replaced and tested.

Subsequent

1. The Performance Testing frequency was changed to require valves FDW-315 and 316 to be stroked tested quarterly without an exception as to the Unit status.

Planned

1. Enhance the Post Trip Review process as necessary, specifically addressing the verification of Emergency Feedwater Flow.

SAFETY ANALYSIS

The purpose of the Emergency Feedwater (EFDW) System is to remove decay heat and cool down the Reactor Coolant System (RCS), in the event that Main Feedwater (MFDW) is unavailable. This system is composed of three EFDW pumps supplying two independent trains, with a control valve present in each train to throttle flow. Each unit has the ability of cross-connecting to either of the other two units if necessary. Two of the EFDW pumps are motor-driven while the third is turbine driven. The accident analyses in the Final Safety Analysis Report (FSAR) only credit EFDW flow from one pump to one steam generator (SG). Thus, any one of these pumps is capable of providing adequate flow to remove RCS heat from any initial power condition. All three pumps receive a start signal on low Main Feedwater (MFDW) header pressure, low turbine oil pressure, or low steam generator level. In the event of a single failure, adequate redundancy is present to assure that the EFDW system will function as designed.

In the event that one of the control valves is inoperable while in the automatic control mode, as was the case with 1FDW-315, a single failure in the other train (1FDW-316) could isolate all EFDW flow to the SGs. This would prevent the EFDW System from performing its intended safety function as assumed in the FSAR accident analyses. However, the operators have the ability to switch control of these valves into manual. Testing of these valves prior to unit start-up was performed in the manual mode. The results of these tests showed that the valves opened as required. The

Emergency Operating Procedure (EOP) instructs the operator to take manual control of these valves in the event that no flow is indicated in the EFDW header(s). Thus, during the time period that 1FDW-315 was inoperable in the automatic mode, operator action could have restored feedwater to the steam generator, even in the event of a single failure in the other train.

In the event that 1FDW-315 was inoperable in both the automatic and manual modes and a single failure in the other train occurred, other means of RCS heat removal are available. The EOP directs the operators to initiate High Pressure Injection (HPI) [EIIS:BG] feed and bleed cooling upon a loss of all primary-to-secondary heat transfer. Adequate time is available between the initiation of a total loss of feedwater event and the time at which feed and bleed begins such that no core damage would occur. This manner of RCS heat removal can be used until MFDW or EFDW flow is restored. If the EOP is followed properly, feed and bleed cooling is capable of removing decay heat and preventing core damage.

In the absence of MFDW and EFDW, an alternative method of heat removal to HPI feed and bleed is the use of the Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) [EIIS:BA] pump. The design purpose of this pump is to supply secondary inventory at flow rates as high as 500 gpm to each unit during SSF event. An SSF scenario can result in a loss of MFDW and EFDW, as well as other safety systems. Flow from the ASW pump enters the EFDW System downstream of control valves FDW-315 and FDW-316. Analyses have been performed to verify that sufficient time is available for an operator to line this system up before any core damage would occur.

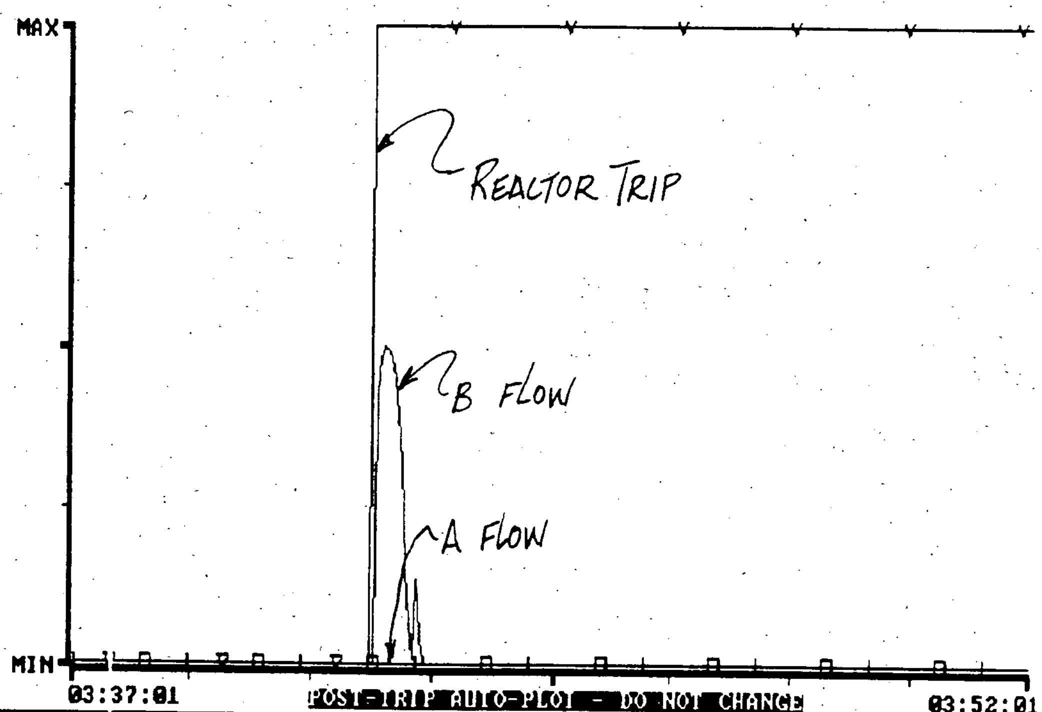
Although the potential existed for the automatic control of the EFDW system to be inoperable, assuming a single failure, adequate means of RCS heat removal were available through the use of operator action to restore EFDW flow, HPI feed and bleed cooling, or use of the SSF ASW pump. Each of these alternate methods of decay heat removal would have been successful in preventing core damage. Therefore, this event did not result in a significant risk to the health and safety of the public.

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	0 5 0 0 0 2 6 9	9 2	— 0 0 5	— 0 0	0 8 OF 0 8

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

ATTACHMENT A



STY	TAG	MIN	MAX	EGU	DESCRIPTOR	NODE
+	A5049	0.00	1200.0	GPM	EFDW FLOW A	OC1030
□	A5050	0.00	1200.0	GPM	EFDW FLOW B	OC1030
▽	D5002	FALSE	TRUE		REACTOR TRIP	OC1030

03:44:31.90 05-08-92