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FACIL:50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410
AUTH.NAME AUTHOR AFFILIATION
JENKINS,R.E. Niagara Mohawk Power Corp.
WILLIS,J.L. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-051-01:on 870813,shutdown cooling sys isolated & Tech Specs 3.4.9.2 exceeded.Caused by equipment failure,personnel error & procedural & design deficiencies.Shutdown cooling sys manually restored.W/880822 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 11
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-1 LA	1	1	PD1-1 PD	1	1
	HAUGHEY,M	1	1	BENEDICT,R	1	1
INTERNAL:	ACRS MICHELSON	1	1	ACRS MOELLER	2	2
	ACRS WYLIE	1	1	AEOD/DOA	1	1
	AEOD/DSP/NAS	1	1	AEOD/DSP/ROAB	2	2
	AEOD/DSP/TPAB	1	1	ARM/DCTS/DAB	1	1
	DEDRO	1	1	NRR/DEST/ADS 7E	1	0
	NRR/DEST/CEB 8H	1	1	NRR/DEST/ESB 8D	1	1
	NRR/DEST/ICSB 7	1	1	NRR/DEST/MEB 9H	1	1
	NRR/DEST/MTB 9H	1	1	NRR/DEST/PSB 8D	1	1
	NRR/DEST/RSB 8E	1	1	NRR/DEST/SGB 8D	1	1
	NRR/DLPQ/HFB 10	1	1	NRR/DLPQ/QAB 10	1	1
	NRR/DOEA/EAB 11	1	1	NRR/DREP/RAB 10	1	1
	NRR/DREP/RPB 10	2	2	NRR/DRIS/SIB 9A	1	1
	NUDOCS-ABSTRACT	1	1	REG FILE 02	1	1
	RES TELFORD,J	1	1	RES/DSIR DEPY	1	1
	RES/DSIR/EIB	1	1	RGN1 FILE 01	1	1
EXTERNAL:	EG&G WILLIAMS,S	4	4	FORD BLDG HOY,A	1	1
	H ST LOBBY WARD	1	1	LPDR	1	1
	NRC PDR	1	1	NSIC HARRIS,J	1	1
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Note: 2nd Distribution due to incorrect Rids Code

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NRC Form 366
(9-83)

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

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 ~~410~~	1 OF ~~08~~

TITLE (4) Shutdown Cooling Isolation and LCO Exceeded due to Design Deficiency and a Lack of Support for Pre-Job Review

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)
08	13	87	87	051	01	08	22	88	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 4
POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[X] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Robert E. Jenkins, Assistant Supervisor Technical Support	315	349-4220

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	ED	52	G080	N					

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)

While in cold shutdown (Mode 4) on August 13, 1987 at 0323 hours, the Shutdown Cooling (SDC) system isolated during performance of the Electrical Protection Assemblies (EPA) calibration surveillance procedure. When EPA 1A was tripped per the surveillance, the inboard SDC valve isolated and the running pump tripped. This was an unanticipated result, since tripping a single power supply normally would isolate either inboard (Division 2) or outboard (Division 1) valves. Upon attempting to reset EPA 1A, SDC could not be restored within the one hour limit per Technical Specifications. The causes of the event were equipment failure, personnel error, procedural deficiency, lack of training, design deficiency and lack of support for pre-job review.

Immediate corrective actions were to monitor coolant temperatures and to manually restore normal SDC by 0525 hours. Normal SDC was lost for approximately two hours, but coolant temperatures remained within the limit for cold shutdown. The EPA breaker was replaced, and the half scram condition and isolations were reset. Further corrective actions include procedural revisions, training for operators, and future EPA modifications.

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1/1

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

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	PAGE (3)		
Nine Mile Point Unit 2	0 5 0 0 0 410	87	—	051	—	01	02	OF	08

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

I. DESCRIPTION OF EVENT

On August 13, 1987 at 0323 hours the Shutdown Cooling (SDC) System isolated during performance of the Reactor Protection System (RPS) Electrical Protection Assemblies (EPA) calibration surveillance procedure. The normal SDC system could not be restored within the one hour limit established in Technical Specifications (TS). Thus, the Limiting Conditions of Operation (LCO) for the SDC system were exceeded, resulting in a violation of TS 3.4.9.2. Prior to the event, the unit was in cold shutdown (Mode 4), with SDC Loop B operating. Reactor vessel pressure was 0 psig and coolant temperature was 130°F.

Refer to Attachment 1 for the Residual Heat Removal System (RHR/RHS)-SDC subsystem configuration. Refer to Attachment 2 for RPS logic power supply configuration.

At 0322 hours, EPA 1A was tripped by Niagara Mohawk Power Corporation (NMPC) Meter and Test (M&T) personnel in accordance with the surveillance procedure to perform the necessary channel calibrations. The expected Division 1 half scram and primary containment isolations (outboard containment isolation valves) were received. However, an unexpected result was the receipt of a Division 2 SDC isolation. Consequently, the inboard SDC suction isolation (2RHS*MOV112) and SDC Loop B injection (2RHS*MOV40B) valves closed. The other Division 2 SDC valves were already closed. Subsequently, the running SDC pump (2RHS*P1B) tripped due to the suction valve closure, and normal SDC was interrupted at 0323 hours.

Prior to tripping the EPA, NMPC licensed control room operators and the Station Shift Supervisor (SSS) made a conscientious effort to take the necessary measures to prevent isolation of systems such as SDC, Reactor Water Cleanup (WCS) and normal Reactor Building Ventilation (HVR). In addition, measures were taken to prevent auto initiation of Engineered Safety Features (ESF) systems such as Standby Gas Treatment (GTS). Except for the Main Steam Isolation Valves (MSIVs), the Nuclear Steam Supply Shutoff System (NS^4) isolation circuits are designed to isolate one of the two isolation valves on a loss of either divisional power supply (i.e. Division 1 isolates outboards and Division 2 isolates inboards). However, the SDC isolation circuit is also an exception to this general design. A loss of any divisional power supply will result in a complete isolation of the SDC system (both inboard and outboard valves). The purpose for this unique design feature is to protect the SDC system from overpressurization. Due to procedural deficiencies and a lack of knowledge of this unique feature of the SDC isolation circuit, the operators' measures to prevent SDC isolations did not include preventing closure of 2RHS*MOV112 or 40B. Thus, SDC isolated when the EPA was tripped.

Upon receiving the unexpected SDC isolation, the SSS instructed M&T personnel to reset the EPA output breaker so that power could be restored to the bus and measures taken to prevent SDC from isolating again. Plans were to proceed with the required calibration, once all necessary preventative measures were taken and normal SDC restored.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	—	051	—	01	03 OF 08

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

However, upon attempting to reset the EPA output breaker, the breaker resetting mechanism failed. At this time the SSS allowed M&T personnel to troubleshoot the breaker in an attempt to correct the failure and reset it. At approximately 0400 hours, M&T personnel indicated that the breaker could not be fixed and that a replacement would be necessary. Consequently, the half scram condition and the NS^4 isolations could not be reset, and normal SDC could not be restored via remote operation.

Immediate corrective actions were to monitor reactor coolant temperatures per TS 3.4.9.2 and to re-establish normal SDC by manually opening 2RHS*MOV112 and 40B and restarting the tripped SDC pump. In addition, an emergency work request was generated to replace the defective EPA output breaker. Operators also began raising reactor water level in an attempt to improve natural circulation.

An alternate decay heat removal method was in operation throughout the event via WCS system. However, an alternate coolant circulation method (i.e. forced circulation) could not be established within the one hour limit. Therefore, by 0423 hours the LCO for the SDC system was exceeded.

At 0525 hours, normal SDC was returned to service. Coolant temperature rose 15°F during the two-hour period but remained within the cold shutdown limits throughout the event.

Subsequently, the EPA breaker was replaced and reset and the half scram and NS^4 isolations were reset at 1508 hours.

II. CAUSE OF EVENT

The immediate cause of the event was the failed EPA output breaker resetting mechanism. The intermediate causes of the event were personnel error, procedural deficiency and lack of training. The root causes were design deficiency and a lack of support for pre-job reviews.

The LCO for SDC capability would not have been exceeded if the EPA output breaker had not malfunctioned. Therefore, the immediate cause was equipment failure. The cause for the equipment failure is unknown. The vendor was requested to perform a failure analysis. However, the vendor recommended that the failure analysis not be performed mainly because the breaker was obsolete and a different design unit is now available. Since the EPA breakers will be replaced with this newer design breaker, Engineering concluded that a failure analysis would not be required in this case.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	051	01	04 OF 08

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

The intermediate causes were personnel error, procedural deficiency and lack of training. The SSS did not initiate timely corrective action to restore normal SDC prior to exceeding the LCO. Troubleshooting activities should have been conducted concurrent with normal SDC restoration activities. Furthermore, the EPA calibration surveillance procedure had an inadequate plant impact statement. Although the plant impact stated that the Residual Heat Removal System would isolate on a loss of an EPA, it did not specify that both divisional valves would isolate from a loss of a single division. This particular design feature was not discussed in training or any operating procedure. Thus, the operators relied on their knowledge and the plant impact statement in the procedure, in addition to reviewing drawings, to determine preventative measures to be taken. Therefore, better training and a more detailed plant impact statement were necessary to alert operators of this unique design feature not common to other systems.

A root cause of the event was a lack of support of Operations personnel for adequate pre-job review. An unnecessary burden was placed upon Operations personnel on shift to correctly determine plant impact through several hours of drawing reviews in order to prepare for the EPA surveillance. This approach increased the possibility of errors in determining plant impact.

Another root cause is the inability to test the EPAs without power interruptions. If this design feature existed, this event would not have occurred. A loss of power to a single power supply of the RPS/NS[4] logic results in a half scram and in numerous Engineered Safety Features (ESF) actuations such as primary containment isolation, secondary containment isolation and automatic initiation of GTS.

III. ANALYSIS OF EVENT

This event had no adverse safety consequences. Normal SDC capability was lost and an alternate method of forced coolant circulation could not be established for approximately two hours. Coolant temperature rose 15°F during that period but remained within the limit for cold shutdown throughout the event. An alternate means of decay heat removal was available via the WCS system. The WCS heat exchangers were able to handle existing decay heat loads and maintain coolant temperatures within the TS limit. A loss of normal decay heat removal methods is a design basis condition mitigated by alternate decay heat removal methods described in FSAR Section 15.2.9, "Failure of RHR Shutdown Cooling", and in normal and emergency operation procedures.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	—	051	—	01	05 OF 08

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

The inadvertent failure of the EPA output breaker resetting mechanism or an unanticipated trip of an EPA has the potential to create a serious transient during power operation. The loss of a single power supply to RPS/NS4 logic has the following plant impact:

1. half scram via the RPS
2. secondary containment isolation
3. auto initiation of the emergency reactor building ventilation recirculation units and GTS
4. isolation of outboard primary containment isolation valves (Groups 2, 4, 6, 8 and 9 and main steam line drains) via NS4
5. isolation of inboard and outboard RHR-SDC system valves via NS4

The consequences of these events are previously analyzed events that are bounded by the "LOCA Inside Primary Containment" spectrum of events (FSAR 15.6.5) and "Inadvertent HPCS Startup" (FSAR 15.5.1).

IV. CORRECTIVE ACTIONS

Immediate corrective actions were to monitor reactor coolant temperatures per TS 3.4.9.2 and to re-establish normal SDC by manually opening 2RHS*MOV112 and 40B and restarting the tripped SDC pump. In addition, an emergency work request was generated to replace the defective EPA output breaker. At 0525 hours, normal SDC was returned to service. Subsequently, the EPA breaker was replaced and reset and the half scram and NS4 isolations were reset at 1508 hours. Further corrective actions are as follows:

1. The plant impact statements in both the six month and eighteen month EPA surveillance procedures were revised to aid the SSS and control room operators in adequately assessing the plant impact for performance of these procedures. In addition, specific work control guidelines for Operations have been developed and incorporated into these procedures to further assist in pre-job planning.

2. The Superintendent of Operations has advised Station Shift Supervisors that alternate contingency actions should be immediately initiated in parallel with normal methods of returning equipment to service, even if it appears that the problem could be resolved in a short period of time.

3. A Training Modification Request (TMR) was initiated to train Operations personnel on the unique design feature of the SDC isolation circuit to assist in preventing recurrence of this event. Training was completed in November of 1987.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	— 051	— 01	06 OF 08

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

4. The newer design breakers will be purchased and installed in the RPS/NS4 logic power bus EPAs during the next scheduled outage.

5. NMPC Engineering is presently evaluating various options that will improve testability of EPAs and enhance reliability. Since procedural and other administrative controls have been implemented to prevent recurrence of this event, design changes are not being implemented at this time. However, design changes are currently being considered as a long term corrective action.

V. ADDITIONAL INFORMATION

A. Components referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Electrical Protection Assemblies (EPA)	92	ED
Reactor Protection System	N/A	JC
Nuclear Steam Supply Shutoff System (NS4)	N/A	JM
Residual Heat Removal System (RHR/RHS)	N/A	BO
Shutdown Cooling System (SDC)	N/A	BO
Reactor Water Cleanup System (RWCU/WCS)	N/A	CE
Control Rod Drive System (CRD)	N/A	AA
Standby Gas Treatment System (GTS)	N/A	BH
Reactor Building Ventilation System (HVR)	N/A	VA
Pump	P	BO
Isolation Valve	ISV	JM
Circuit Breaker	52	ED

B. Previous Similar Events - None

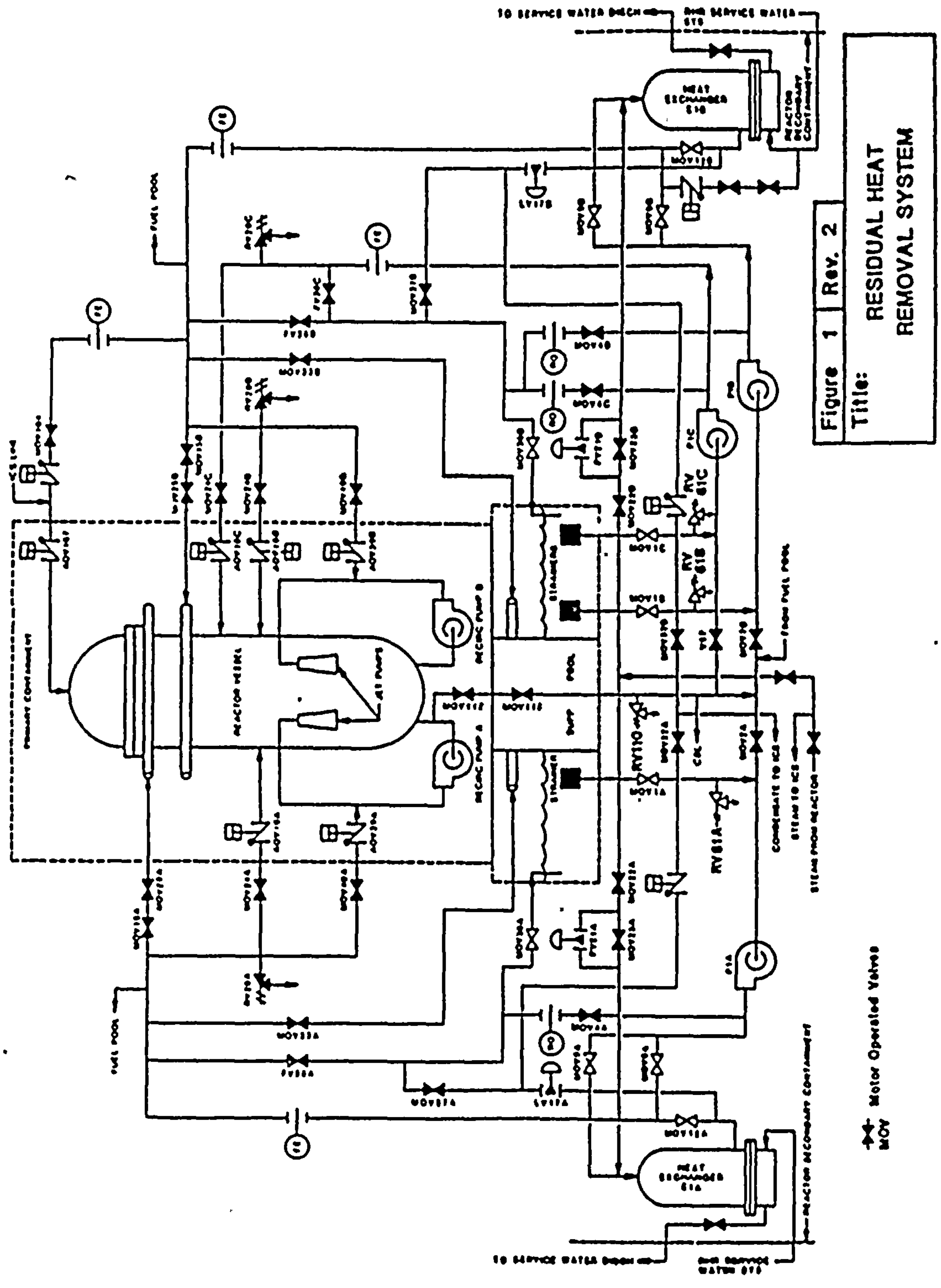
C. Failed Components - General Electric Type TFJ-175A Molded Case Circuit Breaker; Part No. 184C4494P001

*U.S. GPO 1986-0-624-538/455

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	— 051	— 01	07 OF 08

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

Attachment 1



NRC Form 366A (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

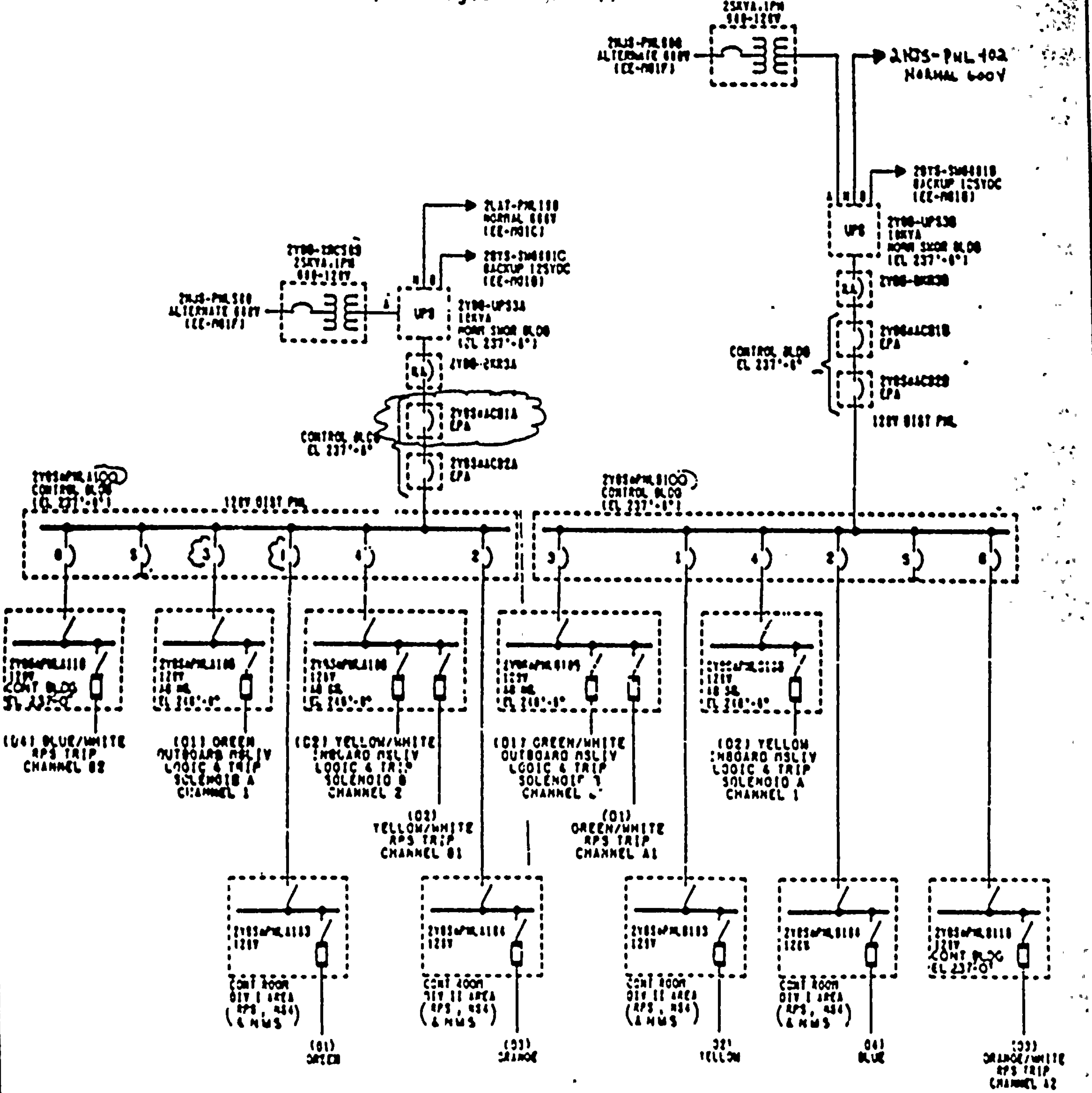
U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 2	0 5 0 0 0 410	87	—	051	—	01	08 OF 08

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

ATTACHMENT 2

RPS/NS4 Logic Power Supplies



NIAGARA MOHAWK

NMP 37830

NINE MILE POINT—UNIT 2/P.O BOX 63, LYCOMING, NY 13093/TELEPHONE (315) 343-2110

August 22, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 87-51, Revision 1

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 87-51 Revision 1 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (i) (B), "Any operation or condition prohibited by the plant's Technical Specifications" and in accordance with 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

10 CFR 50.72 reports were made on August 13, 1987 at 0518 and 0529 hours.

This report was completed in the format designated in NUREG-1022, Supplement No. 2, dated September 1985.

Very truly yours,

J. L. Willis
General Superintendent
Nuclear Generation

JLW/PB/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

IE22
1/1

NIAGARA MOHAWK

NINE MILE POINT—UNIT 2/P.O. BOX 63, LYCOMING, NY 13093/TELEPHONE (315) 343-2110

NMP 37830

August 22, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 87-51, Revision 1

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 87-51 Revision 1 — Is being submitted in accordance with 10 CFR 50.73 (a) (2) (i) (B), "Any operation or condition prohibited by the plant's Technical Specifications" and in accordance with 10 CFR 50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

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