

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8712020418 DOC. DATE: 87/11/25 NOTARIZED: NO DOCKET #
 FACIL: 50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
 AUTH. NAME AUTHOR AFFILIATION
 MAZZAFERRO, P. A. Niagara Mohawk Power Corp.
 LEMPGES, T. E. Niagara Mohawk Power Corp.
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 87-021-00: on 871027, reactor recirculation loop taken from unisolated to fully isolated condition. Caused by personnel error. Consensus will be reached w/all involved parties on all future verbal interpretations. W/871125 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
 TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL
	PD1-1 LA	1 1	PD1-1 PD	1 1
	BENEDICT, R	1 1	HAUGHEY, M	1 1
INTERNAL:	ACRS MICHELSON	1 1	ACRS MOELLER	2 2
	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	1 0	NRR/DEST/CEB	1 1
	NRR/DEST/ELB	1 1	NRR/DEST/ICSB	1 1
	NRR/DEST/MEB	1 1	NRR/DEST/MTB	1 1
	NRR/DEST/PSB	1 1	NRR/DEST/RSB	1 1
	NRR/DEST/SGB	1 1	NRR/DLPQ/HFB	1 1
	NRR/DLPQ/QAB	1 1	NRR/DOEA/EAB	1 1
	NRR/DREP/RAB	1 1	NRR/DREP/RPB	2 2
	NRR/DRIS/SIB	1 1	NRR/PMAS/ILRB	1 1
	<u>REG FILE</u> 02	1 1	RES DEPY GI	1 1
	RES TELFORD, J	1 1	RES/DE/EIB	1 1
	RGN1 FILE 01	1 1		
EXTERNAL:	EG&G GROH, M	5 5	H ST LOBBY WARD	1 1
	LPDR	1 1	NRC PDR	1 1
	NSIC HARRIS, J	1 1	NSIC MAYS, G	1 1



LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Nine Mile Point Unit 1	DOCKET NUMBER (2) 05000220	PAGE (3) 1 OF 015
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TITLE (4) **Failure To Reduce Reactor Power Below Technical Specification Limit Prior To Isolating Recirculation Loop**

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)
11	02	78	78	02	10	11	25	87			050000
THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)											

OPERATING MODE (9) N	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10) 0.919	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)	<input checked="" type="checkbox"/> 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)(ix)	

LICENSEE CONTACT FOR THIS LER (12)

NAME Peter A. Mazzaferro, Assistant Supervisor Technical Support	TELEPHONE NUMBER 3115 3149 1-12 1 1910
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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)

YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

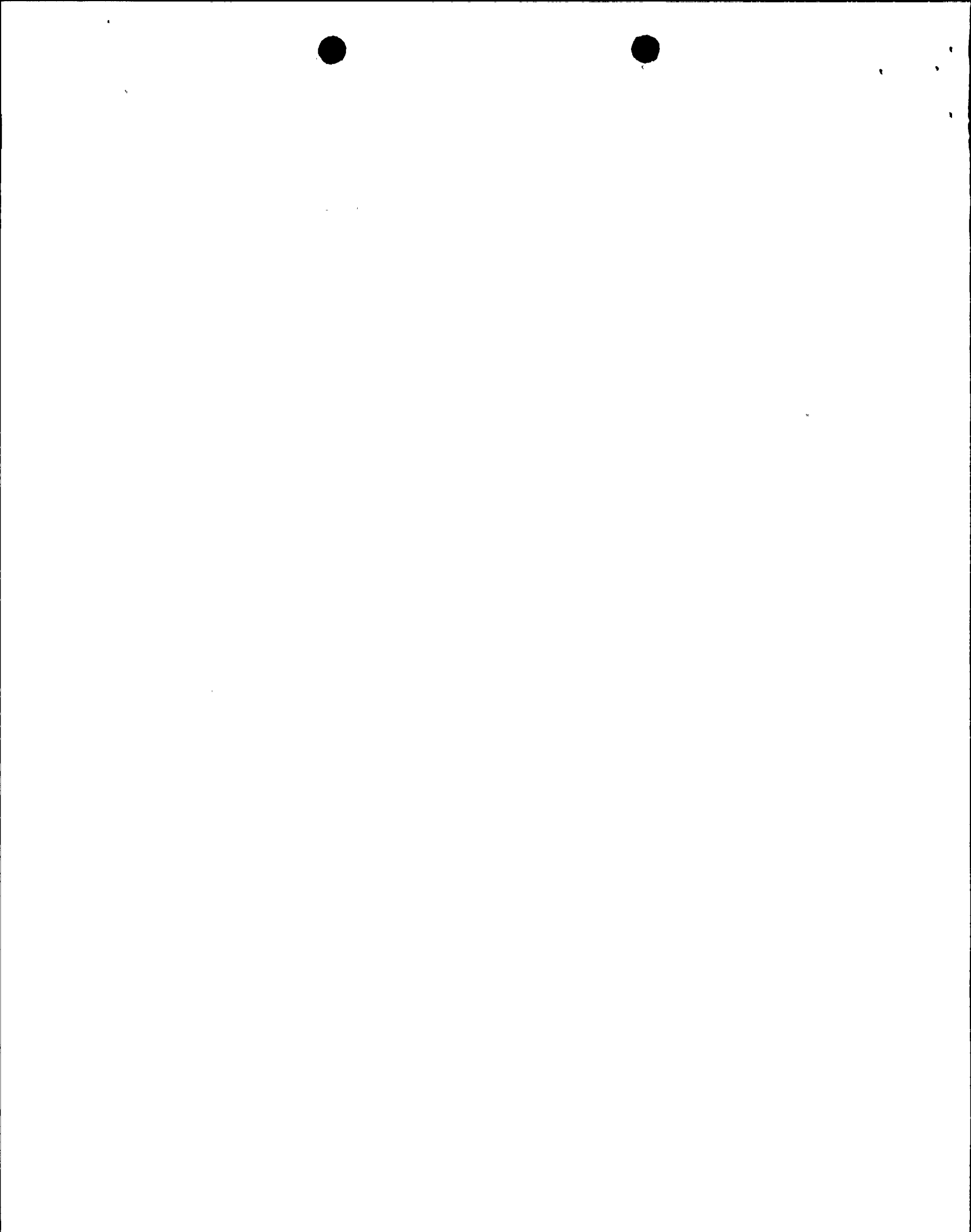
ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

ABSTRACT

On October 27, 1987, with Nine Mile Point Unit 1 operating at 98.5% power, a reactor recirculation loop was taken from the unisolated to the fully isolated condition. The Plant Technical Specifications (Tech. Specs.) state that reactor power shall not exceed 90.5% if neither the fully isolated or unisolated condition can be met to preclude the effects of an inadvertent cold recirculation loop initiation. This condition was not met for 37 minutes while transitioning between the unisolated and isolated loop configuration. A preliminary interpretation of the Tech. Spec. by Station and Licensing staff had concluded that a power reduction below 90.5% was not necessary as a cold water injection could not occur. The recirculation pump in that loop could not be started and the water contained in that loop would not have time to cool appreciably during the transition from the unisolated to fully isolated condition. On October 28, 1987, the written interpretation was received from Licensing which stated that reactor power shall not exceed 90.5% power when transitioning from an unisolated to isolated condition. Based on this interpretation, the activities of the previous day represented a violation of Technical Specification and are, therefore, reportable under the requirements of 10 CFR 50.73.

The root cause of this event is a human performance error by the Station Staff due to a lack of communication.

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FACILITY NAME (1) Nine Mile Point Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 2 2 1 0				LER NUMBER (6)				PAGE (3)		
					YEAR	SEQUENTIAL NUMBER	REVISION NUMBER				
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

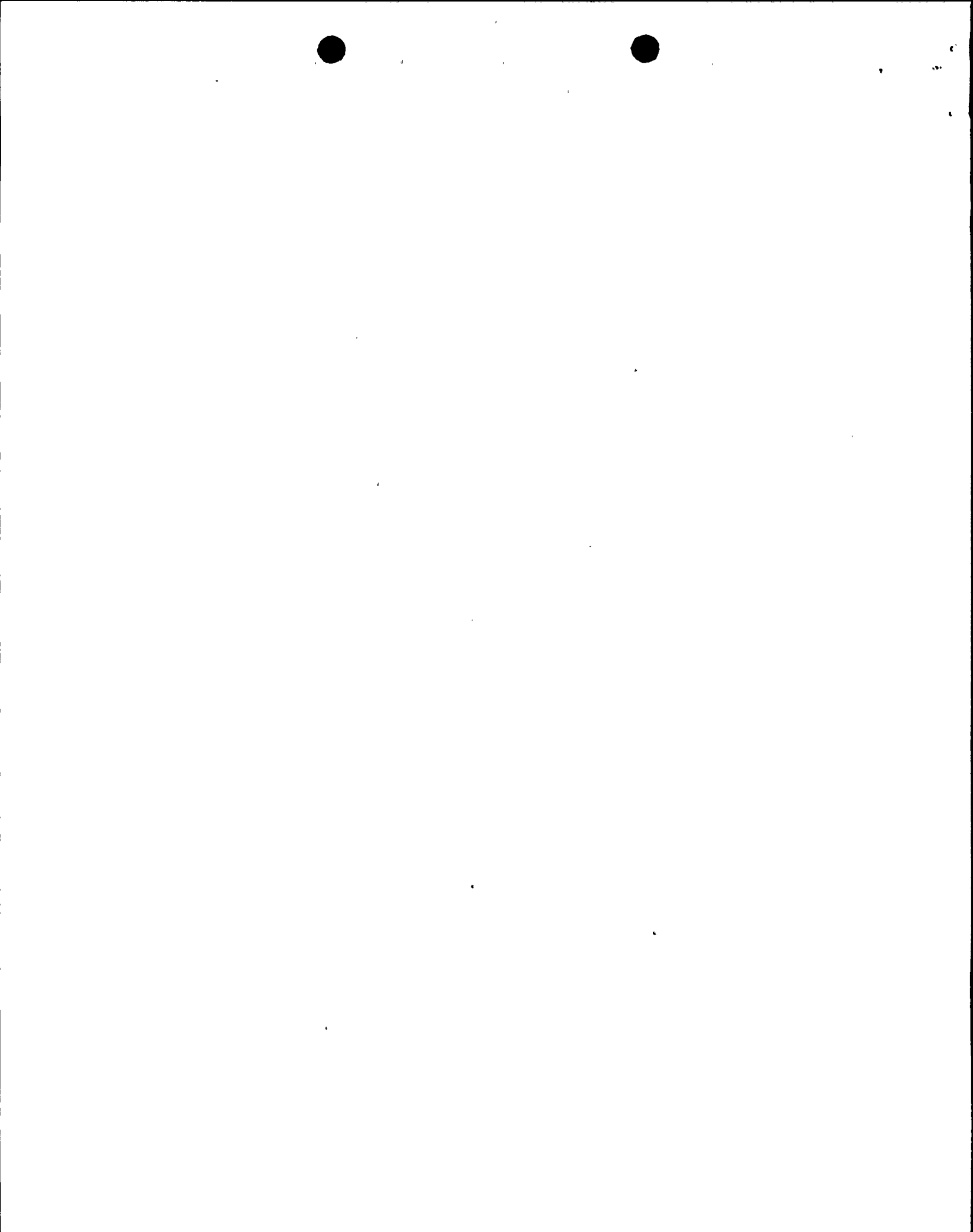
I. DESCRIPTION OF EVENT

At approximately 0845 hours on October 28, 1987, the Nine Mile Point Unit 1 (NMP1) Station Superintendent received a Technical Specification (Tech. Spec.) interpretation from the Licensing Staff stating that reactor power shall not exceed 90.5% when transitioning from an unisolated reactor recirculation (RR) loop to an isolated condition. Since RR loop #14 had been isolated on the previous day with reactor power greater than 90.5%, a Tech. Spec. violation resulted. This event is reportable under 10 CFR 50.73(a)(2)(i)(b) in that the plant was operated in a condition prohibited by Tech. Specs.

Nine Mile Point Unit 1 is a BWR/2 with five RR loops. Each loop consists of a pump with suction, discharge, and discharge bypass valves. The recirculation pump motor speed is controlled via a variable frequency motor-generator set.

On October 26, 1987, NMP1 was preconditioning following a startup which began three days earlier. All five RR pumps were in operation at the time. At 1121 hours, RR pump #14 tripped. Reactor power decreased from approximately 1775 MWt to 1632 MWt due to the pump trip. Operators closed the pump discharge valve to minimize the amount of backflow through the idle loop. The pump suction and discharge bypass valve remained open. Preconditioning was stopped at this time in order to allow for troubleshooting. An equipment markup and station work request were written and troubleshooting commenced at approximately 1345. This work was discontinued at the end of the day shift with no cause for the pump trip identified. Troubleshooting was scheduled to continue the following morning. Reactor power was increased at approximately 1630 hours the same day under preconditioning guidelines.

At the start of the day shift on October 27, 1987, reactor power was at 1826 MWt (98.7%) and troubleshooting had recommenced. Electrical Maintenance personnel had physically disconnected the RR pump #14 motor from its motor generator set in order to meggar the cable. An electrical ground was identified and the decision was made by the Operations Department to isolate RR loop #14. Isolation of an idle RR loop is defined by Tech. Spec. 3.1.7e as; (1) the suction, discharge, and discharge bypass valves are closed and their associated motor breakers locked in the open position, and, (2) the associated pump motor breaker opened and the breaker removed. Full power operation is allowed with one RR loop out of service in either the fully isolated or fully unisolated condition because neither has the potential for a cold water accident due to the inadvertent startup of a cold RR loop. If neither condition can be met, the 90.5% power limitation ensures that the initiation of a cold RR loop would not produce an unsafe transient. Prior to noon on this day, discussions were held between members of the Station Staff regarding the requirement to reduce reactor power while transitioning between



FACILITY NAME (1) Nine Mile Point Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 2 2 0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 7	- 0 2 0	- 0 1 0	0 3	OF 0 5

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

I. DESCRIPTION OF EVENT (Cont'd)

the unisolated and isolated condition. The power supply cable to RR pump #14 remained disconnected following the conclusion of troubleshooting by Electrical Maintenance and as the pump was incapable of being started, there was no possibility of a cold water injection. In addition, as the RR loop had been left unisolated since the pump trip, it remained at near normal operating temperature and would not cool appreciably between the time the remaining loop valves (suction and discharge bypass) were shut and all the associated RR loop valve motor breakers were locked open. Therefore, there was no possibility of a cold water injection from RR loop #14. The Station Staff could not reach a consensus on the need to reduce reactor power to less than 90.5% and a request for Tech. Spec. interpretation from the Licensing Staff was made at approximately 1130 hours. The Licensing Staff reviewed the situation and provided a preliminary interpretation at approximately 1155 hours that a power reduction was not required as that RR pump #14 was incapable of operation and a cold water injection was not possible. At 1305, the suction valve and discharge bypass valve were shut. The equipment markup ensuring that the RR loop was isolated in accordance with the Tech. Spec. requirements was issued at 1342. Thirty seven minutes elapsed from the time the loop was isolated until the Tech. Spec. requirements for the isolated loop were met. During this time period, reactor power was at approximately 98.5% and the isolated loop temperature decreased less than 15°F.

At approximately 1415 hours on the same day (10/27/87), the Licensing Staff telephoned the NMP1 Station Superintendent to report that they had reviewed the situation further and that their prior interpretations may not be correct. They recommended that the RR loop not be isolated until a more thorough evaluation could be completed. By this time, however, the loop had already been isolated. At 0845 hours on October 28, 1987, the Licensing Staff telecopied a written Tech. Spec. interpretation which stated that reactor power could not exceed 90.5% when transitioning between an unisolated and isolated loop. At this time, the events of the previous day were determined to be reportable and a 10 CFR 50.72 notification was made at 0940 and an Occurrence Report was generated in accordance with the Site Administrative Procedure.

II. CAUSE OF THE EVENT

The root cause of this event is a human performance error by the Station Staff. There was a lack of adequate communication among the Station Staff members concerning the question of reducing reactor power prior to isolating RR loop #14. Contributing to this event was an incomplete review by Licensing of the Tech. Spec. prior to providing the Station Staff with an interpretation. They correctly determined that there was no technical reason for reducing reactor power as the intent of the Tech. Spec. was met, but did not do a detailed examination of the wording of the Tech. Spec. until after the preliminary interpretation had already been relayed to the Station Superintendent.



FACILITY NAME (1) Nine Mile Point Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 2 2 0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 7	- 0 2 1	- 0 0	0 4	OF 0 5

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

III. ANALYSIS OF EVENT

There were no adverse safety consequences associated with this event nor was the reactor in an unsafe condition at any time. The restrictions stated in the Tech. Specs. (see Additional Information) are based on precluding a cold water injection transient resulting from an inadvertent startup of an idle RR pump. As described in the Bases for Section 3.1.7 of the Tech. Specs., the concern is with an isolated loop containing cold water, inadvertently being placed back into service, and resulting in a power excursion due to the insertion of positive reactivity. This scenario was not possible during the time frame that RR loop #14 was being isolated for two reasons. The difference in water temperature between the discharge of RR loop #14 and the discharge of the remaining loops was only 15°F. Given the small temperature difference, the amount of positive reactivity would be minimal and the resulting power excursion well within the reactor power protective setpoints. The second reason a cold water injection transient could not occur is due to the fact that the RR pump motor control cable was physically disconnected from its motor-generator set. Therefore, RR pump #14 could not have started even if its control switch was inadvertently turned to start. As such, there were no actual or potential safety consequences associated with this event.

The above described Tech. Spec. violation has potential safety considerations only for reactor power levels greater than 90.5%. (These considerations are discussed above.) Therefore, an assessment of safety consequences at reactor power levels less than 90.5% or other plant operating conditions (i.e. startup, refuel and shutdown) is not applicable.

IV. CORRECTIVE ACTIONS

Corrective action to prevent a similiar event from occurring is that on all future verbal Tech. Spec. interpretations, a consensus will be reached with all the involved parties or the most conservative action will be taken. A formal Tech. Spec. interpretation program will be initiated to catalog and publish written responses to Tech. Spec. questions. A Lessons Learned Transmittal will be routed to the involved Station and Licensing Staff concerning this event.



2

FACILITY NAME (1) Nine Mile Point Unit 1	DOCKET NUMBER (2) 0 5 0 0 0 2 2 0	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 7	- 0 2 1	- 0 1 0	0 5	OF 0 5

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

V. ADDITIONAL INFORMATION

There have been no previous events of this type at Nine Mile Point Unit 1.

Section 3.1.7.e of the NMPI Tech. Specs. read as follows:

"Partial Loop Operation

During power operation, partial loop operation is permitted provided the following conditions are met.

When operating with four recirculation loops in operation and the remaining loop unisolated, the reactor may operate at 100 percent of full licensed power level in accordance with Figure 3.1.7aa and an APLHGR not to exceed 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e.

When operating with four recirculation loops in operation and one loop isolated, the reactor may operate at 100 percent of full licensed power in accordance with Figure 3.1.7aa and an APLHGR not to exceed 98 percent of the limiting values shown in Figures 3.1.7a, 3.1.7b, 3.1.7c, 3.1.7d, and 3.1.7e, provided the following conditions are met for the isolated loop.

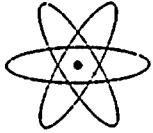
1. Suction valve, discharge valve and discharge bypass valve in the isolated loop shall be in the closed position and the associated motor breakers shall be locked in the open position.
2. Associated pump motor circuit breaker shall be opened and the breaker removed.

If these conditions are not met, core power shall be restricted to 90.5 percent of full licensed power."

Figure 3.1.7aa is the Limiting Power/Flow Line for NMPI. Figures 3.1.7a thru 3.1.7e are graphs of the Maximum Average Planar Linear Heat Generation Rate versus Average Planar Exposure for each fuel type ever used in the NMPI core.



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NIAGARA MOHAWK POWER CORPORATION

NIAGARA  MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

NMP29791

November 25, 1987

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

RE: Docket No. 50-220
LER 87-21

Gentlemen:

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

LER 87-21 Which 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;"

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September, 1985.

Very truly yours,

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/meh

Attachment

cc: William T. Russell
Regional Administrator

IE22
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