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)
Nine Mile Point Unit I	0 5 0 0 0 2 2 0	1 OF 0 6

TITLE (4)
Reactor Scram on Loss of Instrument Air

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)
1 1	0 1	8 5	8 5	— 0 2 1 —	0 0	1 1	2 7	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 8

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)	[ ] 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 G. Randall, Supervisor, Technical Support	3 1 5	3 4 9 - 2 4 4 5

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	
X	L D	D R Y	P 0 5 0	N		X	T A	6 5	G 0 8 0	N	
X	S J	F C U	F 1 3 0	Y		X	S A	T M R	E 0 2 0	Y	

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)

ABSTRACT

During normal operation at 98% power on November 1, 1985, the #12 Feedwater Flow Control Valve malfunctioned causing a 95" high reactor water level condition which tripped the turbine and subsequently caused the reactor to scram. The event was caused by a problem with the instrument air system. Instrument air pressure dropped substantially causing the Feedwater Flow Control Valve lockup circuits to activate as designed. The #12 Feedwater Flow Control Valve did not lock in position upon loss of air, however, because the valve's positioner malfunctioned causing the valve to drift open. High Pressure Coolant Injection was initiated upon turbine trip, but was unable to be manually reset because contacts on the Emergency Trip (limit) Switch failed to change state as designed. To control reactor water level, the #12 Feedwater Pump was locked out and restarted several times, but then failed to restart. The #11 Feedwater Pump remained operable.

Upon turbine trip and Turbine Stop Valve closure a high reactor pressure spike caused five of the six solenoid actuated Electromatic Relief Valves to open. All six Electromatic Relief Valves should have automatically opened. A subsequent investigation revealed that Electromatic Relief Valve #113 had failed to open due to a failure of its solenoid actuator assembly. The remainder of the scram recovery was normal.

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NRC Form 366B (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
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LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

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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 B	R V	D 2 4 5	Y							

NRC FORM 366B (9-83)

TEXT

At approximately 0500 hours on November 1, 1985, preparations were being made to tag out Instrument Air Compressor #13, the normal source of instrument air, for troubleshooting activities on the next shift. When it was taken out of service, the backup instrument air system, compressors #11 and #12, were placed into service. The backup compressors were run unloaded for a time which did not allow for the moisture content of the air to be reduced adequately. Subsequently the filters in Instrument Air Dryer #11 became clogged with water and within a very short time the instrument air pressure dropped to a level which activiated the Feedwater Flow Control Valve loss of air lockup circuits as designed. The #12 air operated Feedwater Flow Control Valve failed to lock in its position as designed, however, because an O-ring in the valve positioner leaked, causing the valve to drift open. This increased feedwater flow to the reactor, causing a high reactor water level turbine trip at the 95" setpoint. With reactor power greater than 45% of rated power, the automatic turbine trip initiated a reactor scram and feedwater flow shifted to the High Pressure Coolant Injection mode as designed. The High Pressure Coolant Injection mode of feedwater was unable to be manually reset by operator action as desired, because a plunger assembly on the turbine's Emergency Governor unit stuck, preventing the Emergency Trip (limit) Switch from resetting. This maintained the High Pressure Coolant Injection relays energized in the turbine control circuitry. Reactor water level was manually controlled by locking out and restarting Feedwater Pump #12 several times. This action resulted in the failure of a timer in the Auxiliary (bearing and gear drive) Oil Pump start circuit, which prevented the #12 Feedwater Pump from being restarted. Feedwater Pump #11 was used to maintain reactor water level throughout the remainder of the scram recovery until Shutdown Cooling was initiated.

Upon turbine trip and Turbine Stop Valve closure a high reactor pressure spike caused the Electromatic Relief Valves to dump excess to the Suppression Chamber to control pressure. Based upon transient analysis from available data including that from the last turbine trip, it appears that all six Electromatic Relief Valves should have automatically opened. The Post Scram Review revealed that one of the six Electromatic Relief Valves, #113, had failed to open during the event. Investigation revealed that the reset springs were jammed along the plunger guide rods of the solenoid actuator of Electromatic Relief Valve #113, causing the valve to remain in the closed position. This failure was attributed to wear of the rod guides. Two other solenoid actuator assemblies on Electromatic Relief Valves #112 and #121 on the same Main Steam Line were found to have similar wear characteristics, however, not as severe.

ASSESSMENT OF SAFETY CONSEQUENCES

There are no potential safety consequences resulting from this event as the engineered safety features and related systems functioned as designed except the following.

1. The temporary failure of the instrument air system.

   A total loss of the instrument air system was evaluated and is described in the FSAR. The evaluation concluded that Nine Mile Point Unit I can withstand this event without adverse consequences to the public health and safety.

2. The malfunction of the #12 Feedwater Flow Control Valve.

   A more conservative senario of continuous maximum feedwater flow into the reactor was evaluated and is described in the FSAR. The evaluation concluded that this event does not pose a safety hazard to the public.

3. The inability to reset the High Pressure Coolant Injection mode of Feedwater after the reactor scram.

   No safety consequences were evident as a result of the stuck plunger assembly on the turbine's Emergency Trip (limit) Switch for the Turbine Emergency Governor. The result is that the single element mode of Feedwater control (either flow or level) remained activated instead of the normal three element mode of Feedwater control. This single element mode of Feedwater control was manually overridden by locking out and restarting a Feedwater pump so that fluctuations between high and low reactor water level could be decreased.

4. The inability to restart the #12 Feedwater Pump due to failure of the Auxiliary Oil Pump's timer.

   The redundant #11 Feedwater pump was available to deliver a maximum of 3800 gpm to the reactor vessel. Also, the Automatic Depressurization and Core Spray Systems were available as a backup so that any safety consequences resulting from a further reduction in reactor water level were within the design basis of the plant.

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

ASSESSMENT OF SAFETY CONSEQUENCES (cont)

5. Failure of one out of six Electromatic Relief Valves.

Pressure Relief Mode: The failure of Electromatic Relief Valve #113 during a worst case pressurization transient; i.e., turbine trip without bypass, would not result in Safety Valve actuation. Any five of the six solenoid-actuated Electromatic Relief Valves opening at 1090 to 1100 psig will keep the maximum vessel pressure below the lowest Safety Valve setting.

Automatic Depressurization System Mode: Only three Electromatic Relief Valves are required to open during a small break LOCA in order to blow down reactor pressure and facilitate initiation of Core Spray Injection, therefore, this event only resulted in a reduction in redundancy.

Therefore, since the minimum number of valves were available, there were no adverse consequences from this event, and the potential consequences are within the design basis of the plant.

CORRECTIVE ACTION

1. Instrument Air Dryer #11 was cleaned and new filter elements were installed and the unit was functionally tested. Also, a bypass valve was repositioned to facilitate drainage of the #11 Instrument Air Dryer. The filter elements are inspected semi-annually per Preventive Maintenance Procedure N1-MPM-SA1. In addition, Instrument Air Compressor #13 was completely overhauled.

2. The two lockup valves in the #12 Feedwater Flow Control Valve Positioner had their O-rings replaced and the unit was functionally tested satisfactorily. Also a section of tubing to the lockup valves was replaced. The redundant #11 Feedwater Flow Control Valve lockup valves were also checked and found to be satisfactory. A preventative maintenance procedure is being developed to preclude recurrence of this type of event.

3. The Turbine Emergency Governor plunger assembly was lubricated and cycled several times and exhibited satisfactory operation.

4. The timer for the Auxiliary Oil Pump to the #12 Feedwater Pump was found to be burned out due to repeated starts of the pump. The timer was replaced and the pump was satisfactorily returned to service.

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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

CORRECTIVE ACTION (cont)

5. The actuator assembly for the #113 solenoid actuated Electromatic Relief Valve was replaced. The other five Electromatic Relief Valves were inspected and it was found that #112 tended to stick in the energized (open) position and that #121 displayed evidence of unacceptable wear. These two actuator assemblies were also replaced. The other three Electromatic Relief Valves exhibited an acceptable amount of wear, but as a preventive maintenance measure, the actuator assemblies are scheduled to be changed out in the 1986 refueling outage. All six Electromatic Relief Valves were recalibrated and satisfactorily tested, and returned to service.

NMP-15113

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD WEST
SYRACUSE, N. Y. 13202

November 27, 1985

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

RE: Docket No. 50-220
LER 85-21

Gentlemen:

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

LER 85-21 Which is being submitted 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). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

A 10 CFR 50.72 report was made at 0631 on 11/1/85.

This Licensee Event Report was completed in the format designated in NUREG-1022, dated September 1983.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/tg
Attachments
cc: Dr. Thomas E. Murley
Regional Administrator

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