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 3

TITLE (4)
Design Calculations Were Found Not In Compliance With FSAR

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)
0 1	1 8	8 6	8 6	— 0 0 1	— 0 0	0 2	1 4	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)							
POWER LEVEL (10)	0 8 2		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)		50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)	X	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 6 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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

ABSTRACT

On January 18, 1986 with the plant operating at 82% power, it was determined that the original calculations for jet impingement loading on the Emergency Condenser Steam Supply Isolation Valves, due to a break inside the guard pipe, were not in compliance with the Nine Mile Point Unit I Final Safety Analysis Report. This was discovered while recalculating the jet impingement loads for the replacement of the Emergency Condenser Piping and Isolation Valves. The new calculations indicated that the actual loading conditions are greater than those incorporated into the original design. The isolation valves are being replaced as part of the upcoming Emergency Condenser Piping Replacement Modification during the 1986 refueling outage.

Initial corrective action included a shutdown of the Nine Mile Point Unit I plant on January 18, 1986 in accordance with Technical Specifications. Also, an engineering investigation to determine if other systems were similarily affected was initiated. This new analysis addressed the Main Steam, Feedwater, Core Spray, Shutdown Cooling, Reactor Water Cleanup and Control Rod Drive Hydraulic Return Systems. These systems have penetrations which utilize a guard pipe configuration similar to that in the Emergency Condenser System. Of these systems only the Emergency Condenser Steam Supply, Main Steam, Feedwater and Reactor Water Cleanup are subject to significant thrust loads due to postulated pipe breaks within the guard pipes. The other systems have normally closed isolation valves and/or check valves which prevent pressurization of the piping within the guard pipes as a result of a postulated break during normal operation.

IE 22
/1

☆U.S.GPO:1984-0-454-481/18759

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

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

TEXT

On January 18, 1986 with the plant operating at 82% power, it was determined that the original calculations for jet impingement loading, due to a pipe break inside the guard pipe on the Emergency Condenser steam supply line were not in compliance with the Nine Mile Point Unit I Final Safety Analysis Report. This condition was discovered during the preliminary engineering for replacement of the Emergency Cooling steam supply piping and isolation valves. The loads on the valve anchors were recalculated and, when compared to those developed during the original plant design, were found to be greater. The major difference between the calculations was the original calculation assumed no pressure buildup within the guard pipe while the new analysis assumes limited venting due to restricted flow conditions while exiting the guard pipe. The resultant increase in load is approximately two times greater than the original jet impingement load. (Reference NMPC - Nine Mile Point Unit I Guard Pipe Configuration Analysis, dated January 1986).

Similar guard pipe configurations exist on the Main Steam, Feedwater, Core Spray, Shutdown Cooling, Reactor Water Cleanup and Control Rod Drive Hydraulic Return Systems. Of these systems only the Emergency Condenser Steam Supply, Main Steam, Feedwater, and Reactor Water Cleanup are subject to significant thrust loads due to postulated pipe breaks within the guard pipes. The remaining systems have normally closed isolation and/or check valves which prevent pressurization of the piping within the guard pipes as a result of a postulated break during normal operation.

ASSESSMENT OF POTENTIAL SAFETY CONSEQUENCES

The assessment of potential safety consequences was based on the leak-before-break concept. A previous analysis performed for a leak-before-break scenario was submitted in conjunction with our response to IE Bulletin 80-11. It included the Emergency Condenser, Main Steam, Feedwater and Reactor Water Cleanup Systems. Although the analysis only considered piping outside of primary containment it was re-evaluated for applicability to piping inside the drywell. The determination concluded that the stresses on the piping inside the drywell were no higher than those outside the drywell. Therefore, the analysis is applicable and demonstrates that for significant through-wall cracks, adequate margin against unstable pipe rupture exists.

In addition, an existing leakage detection system inside the primary containment can detect reactor coolant leakage of less than one gallon per minute. The leak-before-break analysis indicates that the leak rate for a postulated 90 degree circumferential break would exceed one gallon per minute. Therefore, existing mechanisms provide an adequate safety margin even though the new load analysis imposes loads due to pressurization which are greater than those in the original design. As a compensatory measure, a standing order was issued to initiate a reactor shutdown if drywell leakage increases by 1 gallon per minute in a 24 hour period after attaining steady state operation.

These assessments were discussed with the NRC during a meeting on January 23, 1986. As a result of this meeting, a waiver of compliance was issued on January 24, 1986 authorizing the restart of Nine Mile Point Unit I. An amendment to the operating license was also issued on January 28, 1986 allowing continued operation for the remainder of cycle 8.

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

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

CORRECTIVE ACTION

Initial corrective action included a shutdown of the Nine Mile Point Unit I plant on January 18, 1986 in accordance with Technical Specifications. Also, an investigation to determine if other systems were similarily affected was initiated. This new analysis addressed the Main Steam, Feedwater, Core Spray, Shutdown Cooling, Reactor Water Cleanup and Control Rod Drive Hydraulic Return Systems. These systems have penetrations which utilize a guard pipe configuration similar to that in the Emergency Condenser System. Of these systems only the Emergency Condenser Steam Supply, Main Steam, Feedwater and Reactor Water Cleanup are subject to significant thrust loads due to postulated pipe breaks within the guard pipes. The other systems have normally closed isolation valves and/or check valves which prevent pressurization of the piping within the guard pipes as a result of a postulated break during normal operation.

In addition, NMPC will evaluate the need to modify the affected systems and perform any required modifications during the 1986 refueling outage.

NMP-16193

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD WEST
SYRACUSE, N.Y. 13202

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

February 14, 1986

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

RE: Docket No. 50-220
LER 86-01

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(i) and (ii) we hereby submit the following Licensee Event Report:

LER 86-01 Which is being submitted in accordance with 10 CFR 50.73(a)(2)(i)(A): "The completion of any nuclear plant shutdown required by the plant's Technical Specifications," and 10 CFR 50.73(a)(2)(ii)(B): "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant."

A 10 CFR 50.72 notification report was made at 1620 on 1/18/86.

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
cc: Dr. Thomas E. Murley
Regional Administrator

IE22
1/1