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FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220

AUTH.NAME	AUTHOR AFFILIATION
SWEET,K.J.	Niagara Mohawk Power Corp.
CARNS,N.S.	Niagara Mohawk Power Corp.

RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-003-00:on 930306 & 07,RWC inside isolation valve & outside isolation valve respectively,failed TS required LLRT.Caused by improper fit-up of valves internal component. Valve replaced & soft seat assembly installed.W/930406 ltr.

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NINE MILE POINT NUCLEAR STATION/P.O. BOX 32, LYCOMING, N.Y. 13093/TELEPHONE (315) 349-2447

Neil S. "Buzz" Carns
Vice President
Nuclear Generation

April 6, 1993
NMP88342

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

RE: Docket No. 50-220
LER 93-03

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(ii), we are submitting LER 93-03, "Local Leak Rate Tests Exceed Regulatory Limit."

A telephone report of this event was made in accordance with 10 CFR 50.72(b)(2)(i) at 1900 hours on March 7, 1993.

Very truly yours,

Mr. N. S. Carns
Vice President - Nuclear Generation

NSC/JTP/lmc
Attachment

xc: Mr. Thomas T. Martin, Regional Administrator Region I
Mr. Wayne L. Schmidt, Senior Resident Inspector

120006

TE22
1/1

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 1	0 5 0 0 0 2 2 0	1 OF 0 4

TITLE (4)
Local Leak Rate Tests Exceed Regulatory Limit

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)
									N/A	0 5 0 0 0
0 3	0 7	9 3	9 3	— 0 0 3 —	0 0	0 4	0 6	9 3	N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 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)		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	
Mr. Kenneth J. Sweet, Technical Manager	3 1 5	3 4 9 - 2 4 6 2

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
B	C E	I S V	A 3 9 1	Y					
B	C E	I S V	C 6 6 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

X YES *(If yes, complete EXPECTED SUBMISSION DATE)* NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	0 7	0 2	9 3

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

On March 6, 1993 and March 7, 1993, with the Nine Mile Point Nuclear Station Unit 1 (NMP1) in a refueling outage and primary containment not required, valves 33-01R (Reactor Water Clean Up Inside Isolation Valve) and 33-03 (Reactor Water Clean Up Outside Isolation Valve), respectively, failed their Technical Specification required Local Leak Rate Test (LLRT) limit of 5 percent L_{to} (i.e., 13.07 Standard Cubit Feet per Hour [SCFH]). These valves are the isolation valves for primary containment penetration X-154. As a result of the failed LLRTs, the primary containment Technical Specification required leak rate limit of L_t (348.85 SCFH at 22 psig) and the 10CFR50 Appendix J limit of 0.6 L_a (386.45 SCFH at 35 psig) were exceeded. Under this condition, the primary containment is considered inoperable with respect to providing a leakage boundary and represents a degradation of a principal safety barrier.

The cause of valve 33-01R failing its LLRT was improper fit-up of the valve's internal components and incorrect seat tightness when the valve was installed in 1991. The cause of valve 33-03 failing its LLRT was deterioration of the soft seat. A root cause analysis was performed for valve 33-01R and will be performed for valve 33-03. The results of this analysis will be reported in a supplement to this Licensee Event Report.

The immediate corrective action was to declare the valves inoperable. Valves 33-01R and 33-03 were repaired and they subsequently passed their LLRTs.

NRC Form 366
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 1	0 5 0 0 0 2 2 0	9 3	— 0 0 3	— 0 0	0 2 OF 0 4

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

I. DESCRIPTION OF EVENT

On March 6, 1993 and March 7, 1993, with the Nine Mile Point Nuclear Station Unit 1 (NMP1) in a refueling outage and primary containment not required, valves 33-01R (Reactor Water Clean Up Inside Isolation Valve) and 33-03 (Reactor Water Clean Up Outside Isolation Valve), respectively, failed their Technical Specification required Local Leak Rate Test (LLRT) limit of 5 percent L_{to} (i.e., 13.07 Standard Cubit Feet per Hour [SCFH]). These valves are the isolation valves for primary containment penetration X-154. As a result of the failed LLRTs, the primary containment Technical Specification required leak rate limit of L_t (348.85 SCFH at 22 psig) and the 10CFR50 Appendix J limit of 0.6 L_a (386.45 SCFH at 35 psig) were assumed to be exceeded. Under this condition, the primary containment is considered inoperable with respect to providing a leakage boundary and represents a degradation of a principal safety barrier.

During the refueling outage, LLRTs of NMP1 isolation valves were conducted in accordance with the requirements of Nine Mile Point Unit 1's Technical Specifications and 10CFR50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Technical Specifications require that local leak rate tests be performed for penetrations and valves, and that the total leak rate limit is 5 percent L_{to} at a test pressure of 22 psig. L_{to} is defined as the allowable operational leak rate. Technical Specifications also require a primary containment leakage rate limit, L_t, at a test pressure of 22 psig. If these leakage limits are exceeded, corrective actions are specified. Appendix J requires that the combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 L_a. L_a is defined as the maximum allowable leakage rate at the calculated peak containment internal pressure related to the design basis accident. For NMP1, the peak pressure is 35 psig.

The leakage rate during the March 6th and 7th tests could not be quantified with the leak rate monitors that were being used. These monitors were only calibrated up to a maximum of 42.4 SCFH. Valve 33-01R is an Anchor Darling Company 6", double disc gate valve, and valve 33-03 is a Crane Company 6", three-piece design tilting disc check valve.

II. CAUSE OF EVENT

The cause of valve 33-01R failing its leak rate test was improper fit-up of the valve's internal components and incorrect seat tightness when the valve was installed in the Spring of 1991. At that time, the valve passed its LLRT. However, with operation of the valve over time, improper fit-up and incorrect seat tightness caused an inadequate sealing surface inside the valve. The root cause of the improper fit-up and incorrect seat tightness was inadequate written communications in that the vendor manual did not include instructions for fit-up of the stem, wedge and disc assembly and correct seat tightness when installing the valve.

II. CAUSE OF EVENT (cont.)

A contributing cause was the incorrect orientation of the lower wedge as reported in INPO OE 4993, dated December 1991. The vendor did not supply the correct lower wedge orientation instructions in the vendor manual. Upon disassembly of valve 33-01R, the lower wedge was found to be in the incorrect orientation.

The cause of valve 33-03 failing its leak rate test was a soft seat that was deteriorated in several areas. A portion of the soft seat was askew, preventing full closure of the disc assembly. A root cause analysis is being performed for the failure of the soft seat. The analysis will be completed in 60 days. When the analysis is complete, a supplement to this LER will be issued.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a)(2)(ii), as it represents a loss of containment function and therefore, a degradation of a principal safety barrier.

There were no adverse safety consequences associated with this event as NMP1 was in a refueling outage with primary containment integrity not required. The greater than allowable leakage rates were determined as a result of testing and not due to an actual event which would have challenged this function. As such, the health and safety of plant personnel and the general public were not affected.

The Reactor Water Cleanup system is considered a closed system and therefore, any leakage past the isolation valves would be contained within the system. If leakage from the system should develop, it would be into secondary containment and filtered by the emergency ventilation system prior to release to the environment. Therefore, had these conditions existed during power operation and a containment design basis accident occurred, the health and safety of plant personnel and the general public would not have been compromised.

IV. CORRECTIVE ACTIONS

The immediate corrective actions were to declare the valves inoperable, write Deviation Event Reports to document the failures, and issue Work Orders to inspect and repair the valves.

The valve stem, wedge and disc assembly of valve 33-01R were replaced, and proper fit-up and disc tightening was performed according to the vendor representative's latest instructions. Additionally, the lower wedge assembly orientation was corrected. After repair, the valve was local leak rate tested with the resultant leak rate of 3.47 SCFH, which is within the Technical Specification limit of 13.07 SCFH.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Nine Mile Point Unit 1	0 5 0 0 0 2 2 0	9 3	— 0 0 3	— 0 0	0 4 OF 0 4

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

IV. CORRECTIVE ACTIONS (cont.)

In the Fall of 1992, procedure N1-MMP-033-225, "Overhaul of Reactor Clean Up Inside Isolation Valve 33-01R and 33-02R," was revised, based on INPO OE 4993, to incorporate the proper wedge assembly orientation. The vendor manual will be revised by September 1993 to include the vendor's latest instructions for proper fit-up of the valve stem, wedge and disc assembly.

Valve 33-03 had a new disc and soft seat assembly installed. After repair, the valve was local leak rate tested twice, with the resultant leak rates of 1.49 SCFH and 1.67 SCFH, which are within the Technical Specification limit of 13.07 SCFH. In between the tests, the valve was opened and then reclosed.

V. ADDITIONAL INFORMATION

A. Failed components:

Valve 33-01R, Anchor Darling Co., 6", double disc gate valve
Valve 33-03, Crane Co., 6", three-piece design tilting disc check valve

B. Previous similar events:

There were two previous similar events when the Technical Specification allowable leakage rates and the 10 CFR 50 Appendix J limit were exceeded during testing. The details of these events are presented in LERs 88-11 and 91-05. The corrective actions from LER 88-11 would not have prevented this event. The long term corrective action from LER 91-05, a proposed modification for rerouting RWCU system piping to eliminate valves 33-01R and 33-03, may have prevented this event but was not implemented in the 1993 Refuel Outage.

C. Identification of components referred to in this LER:

COMPONENT	IEEE 805 SYSTEM CODE	IEEE 803A FUNCTION
RWCU Inside Isolation Valve 33-01R	CE	ISV
RWCU Outside Isolation Valve 33-03	CE	ISV