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FACIL:50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha   05000410
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RECIP.NAME   RECIPIENT AFFILIATION

SUBJECT: LER 94-007-00:on 941209,actuations of RPS & ESF sys occurred.Caused by poor managerial methods.Implementation of immediate actions for scram & unit brought to cold shutdown. W/950109 ltr.

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January 9, 1995
NMP2L 1521

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 94-07

Gentlemen:

In accordance with 10CFR50.73 (a)(2)(iv), 50.73(a)(2)(i)(A), and 50.73 (a)(2)(i)(B), we are submitting LER 94-07, "Reactor Protection System and Engineered Safety Feature Actuations and a Technical Specification Violation Occurring During Completion of a Technical Specification Required Plant Shutdown."

Telephone reports of this event were made in accordance with 10CFR50.72 (b)(1)(i)(a) at 0946 hours on December 9, 1994 and in accordance with 10CFR50.72 (b)(2)(ii) at 1510 hours on December 9, 1994.

Very truly yours,

KADahlberg

K. A. Dahlberg
Plant Manager - NMP2

KAD/JTP/kab
Attachment

xc: Mr. Thomas T. Martin, Regional Administrator, Region I
Mr. Barry S. Norris, Senior Resident Inspector

IE22
//

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Nine Mile Point Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 4 1 0

PAGE (3): 1 OF 0 8

TITLE (4): RPS and ESF Actuations and a Technical Specification Violation Occurring During Completion of a Technical Specification Required Plant Shutdown

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)
1 2	0 9	9 4	9 4	— 0 0 7 —	0 0	0 1	0 9	9 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 2 2

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: John T. Conway, Operations Manager

TELEPHONE NUMBER: AREA CODE 3 1 5 | 3 4 9 - 2 6 9 8

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	W F	C O N D	I 0 7 5	Yes					

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)

On December 9, 1994 at 1418 hours, while Nine Mile Point Unit 2 (NMP2) was performing a normal reactor shutdown required by Technical Specifications, actuations of the Reactor Protection System (RPS) and Engineered Safety Feature (ESF) System occurred. Specifically, operators initiated a manual reactor scram, from approximately 22 percent rated core thermal power, after the offgas system isolated and condenser vacuum started to decrease. After the scram, reactor vessel water level decreased to level 3, as expected, initiating a Group 4 (Residual Heat Removal) and Group 5 (Shutdown Cooling) isolation signal. Additionally, a violation of Technical Specification Surveillance Requirements occurred when the drywell and suppression chamber atmospheres were purged without first performing required sampling of the suppression chamber. At the time these events occurred, NMP2 was in an Unusual Event due to increased unidentified reactor coolant system leakage in the drywell.

The root cause of the Technical Specification required shutdown, due to reactor coolant system leakage, is poor managerial methods. During the drywell inspection following shutdown, valve 2CSH*HCV120 was found to have a packing leak. This valve was previously scheduled to be repacked during the third refuel outage (Fall 1993), but was removed from the outage work schedule without the risks and consequences being adequately assessed. The root cause of the Technical Specification violation has been determined to be inadequate written communications. The packing leak on valve 2CSH*HCV120 was repaired.

I. DESCRIPTION OF EVENT

On December 9, 1994 at 1418 hours, while Nine Mile Point Unit 2 (NMP2) was performing a normal reactor shutdown required by Technical Specifications, actuations of the Reactor Protection System (RPS) and Engineered Safety Feature (ESF) System occurred. Specifically, operators initiated a manual reactor scram, from approximately 22 percent rated core thermal power, after the offgas system isolated and condenser vacuum started to decrease. After the scram, reactor vessel water level decreased to level 3, as expected, initiating a Group 4 (Residual Heat Removal) and Group 5 (Shutdown Cooling) isolation signal. Additionally, a violation of Technical Specification Surveillance Requirements occurred when the drywell and suppression chamber atmospheres were purged without first performing required sampling of the suppression chamber. At the time these events occurred, NMP2 was in an Unusual Event due to increased unidentified reactor coolant system leakage in the drywell.

On December 9, 1994 at 0835 hours, operators at NMP2 observed an increase in drywell floor drain leakage from 0.9 gpm to 3.5 gpm. Additionally, radiation detectors that monitor the drywell atmosphere alarmed, showing a rising trend in radioactivity. Operators entered Technical Specification Limiting Condition for Operation (LCO) 3.4.3.2.e for a 2 gpm increase in unidentified reactor coolant system leakage in any 24-hour period while in Mode 1. The action statement for this LCO requires that the source of the leakage be identified within 4 hours or be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours. Operators declared an Unusual Event at 0908 hours because of exceeding the Technical Specification leakage limit, and began an orderly shutdown at 0929 hours. During this Technical Specification required shutdown, drywell floor drain leakage continued to increase. At 1202 hours, drywell floor drain leakage was 4.35 gpm and drywell pressure was 0.46 psig.

As reactor power decreased, offgas system operating parameters became erratic. At 24.5% power, both offgas recombiner trains isolated on low temperature. Upon investigation, operators found indications of abnormal operation; specifically, indication of possible high hydrogen concentration and the potential for water intrusion. Operators continued inserting control rods to continue with plant shutdown. When the low power setpoint was reached, the control rod sequence control systems blocked control rod insertions because of a faulty "full out" indication for two control rods. Due to decreasing main condenser vacuum, and an unrecoverable problem in the offgas system, a manual reactor scram was initiated at 1418 hours. The Station Shift Supervisor ordered this scram 10-15 percent higher in power than specified by the normal shutdown procedure. All control rods inserted and, due to shrink in the reactor vessel water level to less than the level 3 setpoint (i.e., 159.3 inches, narrow range indication), a Group 4 isolation (residual heat removal system sample lines and discharge to radwaste line) and Group 5 isolation (residual heat removal system shutdown cooling and reactor head spray line) occurred, as expected. The lowest reactor vessel water level reached

I. DESCRIPTION OF EVENT (cont'd.)

was 158 inches, narrow range indication. Other plant systems responded as expected during the scram.

Following the scram, drywell floor drain leak rate was noted increasing to 5 gpm at 1458 hours. Operators recovered reactor vessel water level, reset the scram and primary containment isolations and continued proceeding to cold shutdown. As reactor pressure decreased, the drywell floor drain leak rate decreased. The Unusual Event was terminated at 1800 hours.

Operators commenced deinerting the drywell and suppression chamber at 1707 hours. The purge was secured at 2215 hours because of loss of the auxiliary boilers and required isolation of the reactor building ventilation systems. Prior to the purge, a gaseous sample was obtained from the drywell, but not the suppression chamber. Technical Specification Surveillance Requirement 4.11.2.1.2 requires sampling and analyzing a representative sample of the containment prior to purging. Samples of the drywell and suppression chamber were obtained and analyzed before recommencing the purge at 0513 hours on December 10, 1994.

The reactor reached cold shutdown at 0215 hours on December 10, 1994.

II. CAUSE OF EVENT

Technical Specification Required Shutdown

The root cause of the Technical Specification required shutdown, due to reactor coolant system leakage, is poor managerial methods. During the drywell inspection following shutdown, valve 2CSH*HCV120 was found to have a packing leak. This valve was previously scheduled to be repacked during the third refuel outage (Fall 1993), but was removed from the outage work schedule without the risks and consequences being adequately assessed.

During the second refuel outage (Spring 1992), the valve packing was found to be leaking. The lantern-ring and remaining packing could not be removed, so only the top four rings of packing were replaced. As a precaution, the valve was backseated. A Work Order was written to repack the valve during the third refuel outage. During the third refuel outage the valve was not repacked, nor was the valve backseated. During the reactor pressure vessel hydro test, the valve packing was found to be leaking, so it's packing was adjusted, resulting in an acceptable leak rate of 15 drops per minute. Thus, the risks and consequences of not repacking this valve nor backseating this valve during the third refuel outage were not adequately assessed.

II. CAUSE OF EVENT (cont'd.)

Reactor Protection System Actuation

The manual reactor scram that was ordered during shutdown was caused by isolation of the offgas system, which resulted in decreasing main condenser vacuum. The cause of the offgas system isolation is water intrusion from the "B" Steam Jet Air Ejector (SJAE) intercondenser into the offgas system recombiner trains.

During the normal plant shutdown, SJAE intercondenser "A" was in service and "B" was out of service with its cooling water inlet valve closed and outlet valve opened. The SJAE intercondensers are cooled by condensate. A feedwater pump was removed from service, resulting in reduced feedwater and condensate flow and increased condensate pressure. This increased pressure likely caused a failure in the "B" SJAE intercondenser, which caused condensate to leak into both "A" and "B" recombiner trains, resulting in low recombiner inlet temperature alarms. The low recombiner inlet temperatures caused both trains of offgas to isolate. The root cause of the failure in the "B" SJAE intercondenser will be determined after the fourth refueling outage (Spring 1995) when a proper inspection and failure analysis can be performed.

Technical Specification Violation

The root cause of the Technical Specification violation has been determined to be inadequate written communications.

Operations Procedure N2-OP-61A, "Primary Containment Ventilation, Purge and Nitrogen System" gives direction to sample the containment prior to purging, but does not specify that both the drywell and suppression chamber must be sampled. Chemistry procedure N2-CSP-CMS-@341, "Containment Purge Evaluation" does not clearly specify that a suppression chamber sample is required for containment purges.

A contributing cause was inadequate managerial methods. The pathway for communications with control room personnel was not clear, and messages were not accurately transmitted to the individual in charge of the purge evolution. During the shutdown, non-shift Operations personnel were assisting control room operators and were answering Chemistry's questions regarding suppression chamber sampling. These conversations should have been referred to the Reactor Operator in charge of purging. Additionally, Chemistry questioned the Station Shift Supervisor (SSS) and he indicated that only the drywell needed to be sampled. Having been told earlier that the purge sample was complete and being analyzed, he believed this question was in regards to a previously requested radioactivity sample of the drywell that was performed as a backup for the containment airborne radioactivity monitors that were in alarm. At the time the SSS was approached, he was very busy with the offgas system problems and the imminent need for a reactor scram.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 50.73 (a)(2)(iv), "any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)," 50.73 (a)(2)(i)(A), "the completion of any nuclear plant shutdown required by the plant's Technical Specifications," and 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications."

The completion of the Technical Specification required shutdown, ultimately as a result of the reactor scram, and subsequent reactor cooldown, complied with the Technical Specification action statement requirements for Reactor Coolant System Operational Leakage. The allowable leakage rates from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. An unidentified leakage increase of greater than 2 gpm within the previous 24 hour period indicates a potential flaw in the reactor coolant pressure boundary, and must be quickly evaluated to determine the source and extent of the leakage. If the leakage rates exceed the values specified or the leakage is located and known to be pressure boundary leakage, the reactor will be shutdown to allow further investigation and corrective action.

The drywell floor drain leak rate had been rising slowly since early November 1994, from approximately 0.2 gpm to as high as 1.3 gpm, but had stabilized at approximately 0.9 gpm. Operators were closely monitoring the leak rate as well as other drywell parameters. The reactor shutdown on December 9, 1994 and subsequent primary containment entry allowed the source of the leak to be identified and corrected. The leak was from the packing on valve 2CSH*HCV120, which is part of the reactor coolant system pressure boundary. After the scram, the leakage increased to a maximum of 5.47 gpm. The initiation of the manual scram during shutdown, at 10-15 percent higher in power than would normally be expected by the normal shutdown procedure, was a conservative action considering the decreasing main condenser vacuum and the unlikely restoration of the offgas system. Scramming the reactor allowed operators to start a mechanical vacuum pump to maintain the main condenser as a heat sink. Other plant systems responded as expected during the scram.

The Primary Containment Isolation System (PCIS) is an Engineered Safety Feature designed to provide an automatic isolation of the process lines penetrating the primary containment. The purpose of the PCIS is to limit the release of radioactive materials to less than that specified by regulatory requirements. However, the Residual Heat Removal (RHR) system valves affected by the Group 4 isolation are not primary containment penetration valves. The isolation provides for the integrity of the RHS "A" and "B" Low Pressure Coolant Injection function. The Group 5 isolation is provided to prevent excessive reactor vessel inventory loss due to a leak in the RHR system. The RHR system is a low pressure system that connects to

III. ANALYSIS OF EVENT (cont'd.)

the reactor coolant system pressure boundary. Low reactor vessel water level, at level 3, initiates both isolations. Both isolations performed their intended safety function.

The offgas system isolation, as a result of low recombiner inlet temperature, is consistent with water intrusion into the offgas recombiners. The offgas system hydrogen alarms, downstream of the recombiners, were likely caused by this water intrusion. The offgas system charcoal adsorbers were evaluated to confirm that charcoal ignition did not occur as a result of the water intrusion. Temperatures at the inlets to the charcoal adsorber tanks did not increase above 75 degrees Fahrenheit and carbon monoxide samples at the adsorber drains were below the acceptance criteria of 20 ppm, indicating no charcoal fires occurred as a result of this event.

Regarding the Technical Specification Surveillance Requirement violation, Chemistry sampled plant stack effluents for noble gas during the purge as required by Technical Specifications. The suppression chamber was purged to the stack via the Standby Gas Treatment System. The stack Gaseous Effluent Monitoring System (GEMS) report during the purge indicated effluent radiation levels which were normal for a shutdown and loss of the offgas system. Based upon the analysis of the stack grab sample obtained during the purge and the stack GEMS report, purge rates did not need adjustment to ensure that dose rates at or beyond the site boundary remained within the limits of Limiting Condition for Operation 3/4.11.2, "Gaseous Effluents - Dose Rate."

Thus, there was no threat to the health and safety of the general public or plant personnel as a result of the event described in this LER.

IV. CORRECTIVE ACTIONS

The immediate corrective action was for the operators to implement immediate actions for the scram in accordance with Operating Procedure N2-OP-101C, "Plant Shutdown." The EOPs were entered to control reactor vessel level and exited as appropriate. The unit was then brought to a cold shutdown condition.

Further corrective actions include:

1. Valve 2CSH*HCV120 was repacked above the lantern-ring, and the valve was backseated to provide an additional barrier to leakage.

2. A Work Order was written to disassemble the valve, remove the lantern-ring and troubleshoot the leakage problem during the fourth refuel outage (Spring 1995).

IV. CORRECTIVE ACTIONS (cont'd.)

3. A Deviation Event Report has been written to evaluate a possible adverse trend regarding the evaluation of equipment deficiencies.

4. Procedures N2-OSP-RPV-@002, "Reactor Pressure Vessel and All Class 1 Systems Leakage Test" and N2-OP-101A, "Plant Startup", the primary containment closeout section, will be revised to specify the level of authority necessary to accept leakage inside containment. This will be completed by March 1, 1995.

5. Offgas intercondenser "A" was inspected for tube leaks with condensate aligned for normal operation to the cooler. No leaks were found. "A" was placed in service and "B" was removed from service and isolated.

6. Repair or replace SJAE intercondenser "B" during the fourth refuel outage. Inspect the intercondenser for failed parts and perform a detailed material failure analysis on those failed parts. This will be completed by July 31, 1995.

7. Chemistry and Operations department's procedures will be revised to ensure that Chemistry sampling and analysis requirements for both drywell and suppression chamber purges are verified complete prior to initiating a purge. This will be completed by January 31, 1995.

8. Operations department procedures will be reviewed for proper interface with support departments. As appropriate, Operations procedures and support department procedures will be revised to ensure that Technical Specification actions performed by support departments are verified complete. This will be completed by January 31, 1996.

9. Administrative procedures will be revised to clarify expectations for interdepartmental communications to and from the main control room involving operational decisions or actions. This will be completed by March 31, 1995.

V. ADDITIONAL INFORMATION

A. Failed components:

Component	-	Steam Jet Air Ejector Intercondenser
Component ID	-	2ARC-E3B
Manufacturer	-	Ingersoll-Rand
Part Number	-	117E-TRBT-46
Description	-	Single pass vertically divided surface condenser

V. ADDITIONAL INFORMATION (cont'd.)

B. Previous similar events:

LER 88-16 describes a similar, but not identical, event where the primary containment was inerted with a nitrogen purge without first obtaining a sample and determining a permissible purge rate. The corrective action was to revise procedure N2-OP-61A to require a sample analysis and permissible purge rate be obtained before purging operations begin. This corrective action would not have prevented the current event.

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM ID
Reactor Protection System	NA	JC
Reactor Vessel	RPV	SB
Primary Containment	NA	NH
Drywell	NA	NH
Suppression Chamber	NA	NH
Primary Containment Isolation System	NA	NH
Radiation Detectors	MON	NH
Drywell Floor Drain	DRN	NH
Valve 2CSH*HCV120	HCV	BJ
Control Rod	ROD	AA
Offgas System	NA	WF
Offgas Recombiner	RCB	WF
Offgas Absorbers	ADS	WF
Main Condenser	COND	SG
SJAE Intercondenser	COND	WF
Residual Heat Removal System	NA	BO
Gaseous Effluent Monitoring System	MON	NA