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FACIL: 50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410

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SUBJECT: LER 87-040-00: on 870703, determined that some assumptions used in calculation for standby gas treatment sys draw down time for secondary containment integrity not consistent w/ plant conditions. Calculation reevaluated. W/870731 ltr.

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
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LICENSEE EVENT REPORT (LER)

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TITLE (4) Secondary Containment Integrity Not Maintained due to Plant Conditions Not Consistent with Assumptions for the Standby Gas Draw Down Calculation

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)
07	03	87	87	040	00	07	31	87	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 000

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)	[X] 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	315	349-2445

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

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH 01 DAY 30 YEAR 88

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

On July 3, 1987, in support of an effort to file a request to increase the Nine Mile Point Unit 2 Technical Specification (TS) allowable service water temperature, it was determined that some of the assumptions used in the calculation for the Standby Gas Treatment (SBGT) system draw down time for secondary containment integrity were not consistent with the current plant conditions. This could have resulted in draw down times in excess of that reviewed and approved in the Safety Evaluation Report, NUREG-1047 Supplement 3.

Immediate corrective actions were to reevaluate the calculation and to impose administrative limits on plant operation. On July 13, 1987, a potentially more limiting scenario for the SBGT draw down time was identified and new administrative limits were imposed.

Corrective actions have been initiated by maintaining the reactor building unit coolers in operation and establishing a minimum temperature differential between reactor building air and service water discharge header temperature of 15°F. A modification has been initiated to automatically start the unit coolers on a Loss-Of-Coolant-Accident (LOCA) signal and to monitor this differential temperature.

Analysis of the SBGT system draw down time is continuing with a computer model which is able to better reflect post-LOCA conditions in the reactor building. Analysis to extend the draw down time from 129 seconds to 6 minutes is also in progress.

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

I. DESCRIPTION OF EVENT

On July 3, 1987, in support of an effort to file a request to increase the Nine Mile Point Unit 2 Technical Specification (TS) allowable service water temperature, it was determined that some of the assumptions used in the original calculation for the Standby Gas Treatment (SBGT) system draw down time for secondary containment integrity were not consistent with the current plant conditions. This could have resulted in draw down times in excess of that reviewed and approved in the Safety Evaluation Report, NUREG-1047 Supplement 3.

Each SBGT subsystem is required to draw down the secondary containment pressure to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 129 seconds following a Loss-Of-Coolant-Accident (LOCA). The calculation used to determine this time requirement assumed that the reactor building unit coolers would be operating at the time of a LOCA. Operation of the unit coolers is required to provide heat removal capability to reduce pressure inside the secondary containment following a LOCA. The SBGT system, by itself, cannot remove secondary containment air at a sufficient rate to establish the 0.25 inch of vacuum water gauge. The calculation also made assumptions which resulted in a 23°F differential between the reactor building (secondary containment) ambient temperature and service water temperature. With the reactor building temperature maintained ≤ 85°F with a unit cooler setpoint of 85°F and a maximum allowable service water temperature of 76°F, at the initiation of a LOCA the unit coolers would not have been in operation and the 23°F temperature differential would not have existed, invalidating the SBGT draw down time analysis.

Initial corrective action was to reevaluate the SBGT draw down time calculation. This analysis reduced the required temperature differential from 23°F to 16°F. Once the differential temperature was determined, administrative controls were imposed to maintain the reactor building air temperature above 85°F to assure continued operation of the reactor building unit coolers and to maintain the temperature differential greater than 16°F.

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On July 13, 1987 the required temperature differential was revised from 16°F to 20°F. The draw down time analysis performed on July 3 assumed a worst case single failure of one diesel generator. A further Engineering review of the SBGT draw down time analysis identified a potentially more limiting worst case single failure. This failure, the failure of a 600 volt electrical bus, would render one division of the safety-related unit coolers and the SBGT system inoperable, while leaving major divisional heat loads operational.

Continued analysis into the SBGT system draw down time with a reduced amount of air inleakage into the reactor building has currently placed operating limits of reactor building temperature and differential temperature of ≥ 85°F and ≥ 15°F, respectively.

There were no components or systems which were inoperable and/or out of service which contributed to the event. No plant system or component failures resulted from the event.

II. CAUSE OF EVENT

The root cause of the event was that the calculation used to determine the SBGT draw down time made non-conservative operational assumptions, which were not converted into operational requirements. The calculation assumed a minimum number of reactor building unit coolers in operation at the time of a LOCA. The calculation also assumed the design maximum allowable temperature for the reactor building of 104°F and a service water temperature of 81°F. This was the basis of the 23°F temperature differential. However, for establishing the most limiting SBGT draw down time, a minimum temperature differential between the reactor building air and service water should have been assumed. The lower the temperature differential, the lower the heat removal capability of the unit coolers. Had a minimum temperature differential been assumed, this reduced heat removal capability would have been noted and either a modification request and/or operational limits could have been imposed.

The Operations Department, however, was not made aware of the assumptions used for the SBGT draw down time calculation. Therefore, the calculation's assumptions to have the unit coolers in operation at the time of a LOCA and a 23°F temperature differential between the reactor building air and the service water has not been maintained during plant operation.

The assumption for the unit coolers to be in operation at the time of a LOCA and the differential temperature assumption were not specifically stated outside the draw down time calculation. These assumptions were not identified during the normal Engineering review process as being operational restrictions and therefore, had not been translated into specific operational requirements.

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III. ANALYSIS OF EVENT

The analysis of the radiological consequences of a LOCA inside primary containment presented in Sections 6.2.3 and 15.6.5 of the Nine Mile Point Unit 2 Final Safety Analysis Report (FSAR) is a very conservative analysis. It follows the methods/assumptions and conditions of Nuclear Regulatory Commission Standard Review Plan (SRP) 15.6.5 (NUREG-800), and Regulatory Guides 1.3 and 1.7. The most restrictive assumption in the analysis is that 100% core noble gas inventory and 25% core halogen inventory are released to the drywell and 50% core halogens are immediately released to the suppression pool. This assumes massive fuel damage. However, this is a very conservative assumption in that the Emergency Core Cooling Systems (ECCS) are designed to actuate in sufficient time, even in the event of the worst single failure, to prevent the maximum fuel cladding temperature from exceeding 2200°F, limit local oxidation of the fuel cladding to 17% of the total cladding thickness before oxidation, limit total hydrogen generated to 1% of the total hypothetical amount which could be generated, maintain the core in a geometry amenable to cooling and maintain the core temperature acceptably low by decay heat removal. Compliance with these requirements assures there would be significant fuel failures and there would be no significant fission product release to the containment (only coolant activity is released).

Section 15.6.5.5.5 of the NMP2 Final Safety Analysis Report (FSAR) discusses a more realistic but still conservative analysis of a LOCA. This analysis assumes only reactor coolant activity (no significant fuel failures) is released to the reactor building for release directly to the environment for the first 129 seconds of the accidents. This results in offsite doses which are only a small fraction of the guidelines established per 10CFR100. If it is assumed that the worst case condition exists, in that the unit coolers never reach the 85°F temperature limit for actuation, then the unit coolers would never cooldown the reactor building to obtain the 0.25 inch of vacuum water gauge pressure. Therefore, unfiltered ground level releases will occur for the duration of the accident. However, 10CFR100 offsite doses will still not be exceeded with no significant fuel failure present. This is based on the fact that since the SBGT system and the elevated release of the radioactivity is conservatively estimated to reduce the dose by a factor of 10,000, multiplying the doses presented in the FSAR for the realisitc analysis of the radiological effects of a LOCA by this factor, the doses would still remain below 10CFR100 guidelines.

Therefore, using a realistic but still conservative analysis of the LOCA accident, offsite doses would remain less than the 10CFR100 guidelines and no significant safety hazard existed.

IV. CORRECTIVE ACTION

For both the July 3 and July 13 events, once the determination was made that the SBGT draw down time calculation was not valid, Site Service Memorandums (SSM) were issued. These SSM's placed administrative controls on the reactor building air temperature, and the temperature differential between the reactor building air temperature and service water temperature. The July 3 SSM set operation limits of ≥ 85°F and ≥ 16°F, respectively. The July 13 SSM set operating limits of ≥ 85°F and ≥ 20°F, respectively.

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Following additional review of the SBGT draw down time calculation, a current minimum temperature differential of ≥ 15°F has been established. This analysis, which combines the most limiting assumptions of both the FSAR and SRP 6.2.3, assumes the scenario of a LOCA with a loss of offsite power and the loss of a Division II 600-volt powerboard. Loss of the 600-volt powerboard results in the loss of the Division II unit coolers and SBGT train, while leaving major heat loads (ECCS pumps/motors) in operation. This analysis also assumes a reduced amount of air inleakage into the reactor building based on additional testing of actual inleakage. The reactor building unit coolers' setpoints have been temporarily lowered to 72°F to maintain the unit coolers in operation. Spare unit coolers, also assumed to be in operation at the time of a LOCA for this analysis, have been valved into service.

Modification N2Y87MX140 will remove the requirement for maintaining the reactor building unit coolers in operation. The SBGT draw down time calculation assumes that the reactor building unit coolers are in operation at the start of the LOCA. Part of Modification N2Y87MX140 will install additional logic to automatically initiate the reactor building unit coolers upon receipt of a LOCA signal. Once the modification is completed, the only temperature restriction for the SBGT draw down time will be the temperature differential between reactor building air and service water. Modification N2Y87MX140 will also install low (19°F) and low/low (16°F) temperature differential alarms. These alarms will provide the operators with early detection that the limiting operational condition is being approached. This early indication will allow time for the operators to either increase reactor building air temperature or to declare the SBGT system inoperable and take the appropriate action statements per TS 3.6.5.3. Per procedure N2-OSP-LOG-S001, "Shift Checks", temperatures are now recorded twice each operating shift to verify reactor building air temperature ≥ 85°F and a differential of ≥ 15°F between reactor building air temperature and service water temperature, whenever the reactor coolant is ≥ 200°F.

Investigation into the post-LOCA SBGT draw down time is continuing. A new computer program is being developed to better model the reactor building conditions following a LOCA. The current ANNULUS computer code models the reactor building as a single volume. The new THREE "D" computer code will allow the reactor building to be modeled as multiple volumes. This compartmentalization will better reflect the heat load distribution within the reactor building. Major heat sources may be located in subcompartments, resulting in localized temperatures greater than the average reactor building temperature. The higher localized temperatures will allow for more efficient heat removal by the unit coolers, which should produce a lower differential temperature requirement.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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The continuing investigation will analyze the SBGT draw down time for a scenario which is more limiting than those presented in the FSAR and the SRP. This scenario is a LOCA with no loss of offsite power, the loss of the Division II 600 volt powerboard, Category II lights off and spare unit coolers inservice. With no loss of offsite power additional heat loads would be present, which will be reflected in the required temperature differential for a 129 second draw down time. Analysis on the radiological effects of an extended draw down time is also being performed in an effort to extend the draw down time from 129 seconds to 6 minutes.

Completion of Modification N2Y87MX140 is currently scheduled for August 10, 1987. Completion of the additional analysis and the establishment of the required temperature differential between the reactor building air and service water is currently scheduled for August 15, 1987. Completion of the analysis for the radiological effects of an extended draw down time is scheduled for late 1987. Upon completion of this analysis a supplemental report will be submitted to present the final resolution for the SBGT system draw down time. This supplemental report will be submitted by January 30, 1988.

V. ADDITIONAL INFORMATION

Identification of Components Referred to in this LER

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
Standby Gas Treatment	N/A	BH
Unit Cooler	CLR	VA
Service Water	N/A	KE
Temperature Alarm	TA	N/A

There have been no previous similar events at NMP2.

NMP26439

NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

301 PLAINFIELD ROAD
SYRACUSE, NY 13212

THOMAS E. LEMPGES
VICE PRESIDENT—NUCLEAR GENERATION

July 31, 1987

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

RE: Docket No. 50-410
LER 87-40

Gentlemen:

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

LER 87-40 Is being submitted in accordance with 10 CFR 50.73 (a) (2) (v), "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident."

A 10 CFR 50.72 (b) (2) (iii) report was made at 1325 hours on July 3, 1987 and a 10 CFR 50.72 (b) (2) (i) report was made at 1700 hours on July 13, 1987.

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

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President
Nuclear Generation

TEL/JTD/mjd

Attachments

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W. A. Cook

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