# CATEGORY

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ACCESSION NBR:9803130097 DOC.DATE: 98/03/02 NOTARIZED: NO DOCKET #
FACIL:50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410

| AUTH.NAME | AUTHOR AFFILIATION |
|---|---|
| DEAN,R. | Niagara Mohawk Power Corp. |
| DAHLBERG,K.A. | Niagara Mohawk Power Corp. |
| RECIP.NAME | RECIPIENT AFFILIATION |

SUBJECT: LER 98-003-00:on 980129,sys outside design basis due was noted.Caused by inappropriate seismic criteria.Preliminary reviews determined affected sys were functional.W/980302 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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NIAGARA MOHAWK

GENERATION
BUSINESS GROUP

NINE MILE POINT NUCLEAR STATION/LAKE ROAD, P.O. BOX 63, LYCOMING, NEW YORK 13093

March 2, 1998
NMP2L 1759

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

RE: Docket No. 50-410
LER 98-03

Gentlemen:

In accordance with 10CFR50.73 (a)(2)(ii)(B), we are submitting LER 98-03, "Systems Outside the Design Basis Due to Inappropriate Seismic Criteria."

Very truly yours,

K A Dahlberg

Kim A. Dahlberg
Plant Manager - NMP2

KAD/TWP/cmk
Attachment

xc: Mr. H. J. Miller, Regional Administrator, Region I
Mr. B. S. Norris, Senior Resident Inspector
Records Management

1/1
IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

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

# LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| Nine Mile Point Unit 2 | 05000410 | 1 OF 6 |

TITLE (4)
Systems Outside the Design Basis Due to Inappropriate Seismic Criteria

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAMES | DOCKET NUMBER(S) |
| 01 | 29 | 98 | 98 | 03 | 00 | 03 | 02 | 98 | N/A | 05000 |
| | | | | | | | | | N/A | 05000 |

OPERATING MODE (9): 1

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check one or more of the following)* (11)

POWER LEVEL (10): 079

| | | | |
|---|---|---|---|
| ☐ 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) | ☒ 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 |
|---|---|
| R. Dean - Manager Engineering NMP2 | (315) 349-4240 |

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) | | EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|---|---|
| ☐ YES *(If yes, complete EXPECTED SUBMISSION DATE)* | ☒ NO | | | | |

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

On January 29, 1998, Nine Mile Point Unit 2 (NMP2) discovered that temporary lead shielding installations implemented during outages when the plant was in cold shutdown or refueling had been analyzed using Operating Basis Earthquake (OBE) inputs only and incorrect allowable stresses for seismic qualification, which resulted in the plant being outside the design basis. At the time of discovery, NMP2 was in the RUN mode with reactor thermal power at approximately 79 percent. A power ascension was in progress. This deficiency was identified during a review of NRC Information Notice (IN) 97-71, "Inappropriate Use of 10CFR50.59 Regarding Reduced Seismic Criteria for Temporary Conditions".

The root cause of this event is a knowledge deficiency regarding the use of probabilistic arguments for licensing and design basis criteria. Personnel involved with qualifying temporary shielding applications did not understand the allowances for the use of probabilistic arguments.

Preliminary reviews determined that the affected systems were functional. Additional reviews will be completed to confirm this. Currently installed shielding configurations were verified to be acceptable. Instruction was provided on the appropriate Safe Shutdown Earthquake (SSE) criteria and Updated Safety Analysis Report (USAR) allowable stress requirements. Additional training will be provided on the use of probabilistic arguments. Engineering design standards for temporary shielding will be revised. Reviews will be performed to evaluate other uses of probabilistic arguments.

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

## I. DESCRIPTION OF EVENT

On January 29, 1998, Nine Mile Point Unit 2 (NMP2) discovered that temporary lead shielding installations implemented during outages when the plant was in cold shutdown or refueling had been analyzed using Operating Basis Earthquake (OBE) inputs only and incorrect allowable stresses for seismic qualification, which resulted in the plant being outside the design basis. At the time of discovery, NMP2 was in the RUN mode with reactor thermal power at approximately 79 percent. A power ascension was in progress. This deficiency was identified during a review of NRC Information Notice (IN) 97-71, "Inappropriate Use of 10CFR50.59 Regarding Reduced Seismic Criteria for Temporary Conditions". The qualification of temporary lead shielding using OBE inputs only and incorrect allowable stresses for seismic qualification was similar to the issue described in IN 97-71.

Plant procedures and activities for which seismic evaluations are required have been reviewed to determine if reduced seismic criteria is allowed. Procedures for activities involving temporary modifications, scaffold erection, rigging, and seismic review of non-safety related components in safety related areas require the application of seismic criteria in accordance with Updated Safety Analysis Report (USAR) design requirements.

For temporary conditions such as the application of lead shielding on piping or components during outages when the plant is in cold shutdown or refueling (modes 4 or 5), a review of typical calculations showed that piping and supports were analyzed using OBE inputs only. In some instances, due to the low probability of an earthquake occurrence during the short duration of the shielding installation, it was concluded that the allowable stresses for emergency conditions instead of upset conditions could be used in the calculations. Since USAR Table 3.9A-2 requires using the lower allowable stresses for OBE loading (upset condition), this method was not in compliance with the NMP2 licensing basis. This error was carried over into an engineering design standard that was therefore also incorrect.

The licensing basis does not provide any allowances for using only OBE during modes 4 or 5. 10CFR50.59 preliminary evaluations or applicability reviews were performed for the lead shielding requests, but did not identify the discrepancy.

There are currently no operability or safety concerns as a result of this deviation since the temporary lead shielding was removed prior to drywell close-out and plant startup. In addition, a limiting seismic event did not occur. Lead shielding which was or is still applied on piping or components during plant operation, in startup or hot shutdown conditions had been properly evaluated for the applicable design conditions, including Safe Shutdown Earthquake (SSE) criteria, and therefore was determined to be acceptable.

The initial review identified that the allowable pipe stresses would have been exceeded for the Residual Heat Removal System (RHS), which placed this system outside the design basis. Based upon preliminary

# I. DESCRIPTION OF EVENT (cont'd)

evaluation of various systems required to be available during plant outages, three other systems were identified which exceeded the allowable stresses for seismic qualification and thus were also outside the design basis. These systems include:

- High Pressure Core Spray System (HPCS)
- Reactor Coolant Recirculation System (RCS)
- Reactor Water Cleanup System (WCS)

# II. CAUSE OF EVENT

The root cause of this event is a knowledge deficiency regarding the use of probabilistic arguments for licensing and design basis criteria. Personnel involved with qualifying temporary shielding applications did not understand the allowances for the use of probabilistic arguments.

A contributing cause is an adequate verification was not performed. The preparers and reviewers of the preliminary evaluations or applicability reviews did not understand the basis of the calculations and did not perform an adequate review of the licensing basis documents.

# III. ANALYSIS OF EVENT

The event is reportable in accordance with 10CFR50.73 (a)(2)(ii)(B), "Any event or condition that resulted in the nuclear power plant being: B) In a condition that was outside the design basis of the plant."

The High Pressure Core Spray System (HPCS) is one of the four Emergency Core Cooling Systems (ECCS) provided to maintain fuel cladding below the temperature limit in 10CFR50.46 in the event of a breach in the Reactor Coolant Pressure Boundary (RCPB) that results in a loss of reactor coolant. HPCS can provide and maintain an adequate coolant inventory inside the Reactor Pressure Vessel (RPV) to limit fuel cladding temperatures in the event of a small break in the RCPB. HPCS operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero pressure. The HPCS cooling decreases vessel pressure to enable the low-pressure cooling systems to function. HPCS is also relied upon as a backup for the Reactor Core Isolation Cooling System (RCIC).

The Reactor Coolant Recirculation System (RCS) consists of two recirculation pump loops external to the RPV. These loops provide the piping path for the driving flow of water to the RPV jet pumps. The variable

## III. ANALYSIS OF EVENT (cont'd)

position hydraulic flow control valve operates in conjunction with a two-speed pump to control reactor power level through the effects of coolant flow rate on moderator void content.

The Residual Heat Removal System (RHS) can remove decay and sensible heat during and after plant shutdown, inject water into the RPV following a Loss of Coolant Accident (LOCA) to reflood the core independently of other core cooling systems, and can remove heat from the primary containment following a LOCA, to limit the increase in primary containment pressure.

The Reactor Water Cleanup System (WCS) can recirculate a portion of reactor coolant through filter demineralizers to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

Although the above systems were determined to be outside the design basis due to the incorrect seismic criteria being applied, preliminary evaluations show that the systems would have remained functional during a postulated SSE event based on the pipe stress allowables of ASME Section III Appendix F. Therefore, no piping failures would have occurred and these systems would have performed their intended functions had they been required. As a result, there were no adverse consequences on the health and safety of the general public or plant personnel.

## IV. CORRECTIVE ACTIONS

1. A verification was performed that lead shielding which was or is still applied on piping or components during plant operation, in startup or hot shutdown conditions had been properly evaluated for the applicable design conditions, including SSE criteria, and therefore was determined to be acceptable.

2. The Engineering Manager and structural design group were instructed to include SSE loading in the piping stress analysis and to use the allowable stresses per USAR requirements. Compliance with NMP2 licensing commitments and USAR requirements were also reemphasized.

3. Preliminary reviews have determined that, based on pipe stresses, the affected systems were functional and would have performed their functions had an SSE occurred while the temporary lead shielding was installed. Additional reviews will evaluate other system components, such as pipe supports, valves, nozzles, etc., to evaluate their impact. These reviews will be finalized and verified by October 30, 1998 to confirm whether or not design basis allowables would have been met and whether the systems would have remained functional.

IV. <u>CORRECTIVE ACTIONS</u> (cont'd)

4. Engineering personnel will be trained on the use of probabilistic arguments and how these arguments apply to the licensing and design basis. This training will be completed by April 30, 1998.

5. Structural engineering design standards for temporary lead shielding will be revised to include:

   - The SSE loading in the piping stress analysis and direction to use the allowable stresses per USAR requirements.

   - A requirement to revise existing calculations to meet the above method when existing calculations are used for qualification of future temporary shielding.

   - A requirement that the applicability review be completed by the calculation preparer, checker and approver rather than other engineering personnel.

   This action will be completed by March 13, 1998. Temporary lead shielding packages which may be issued prior to March 13 will be reviewed to ensure compliance with the correct allowable stresses and criteria described above.

6. Additional reviews will be performed of other engineering programs to ensure that probabilistic arguments are being used correctly. This review will be completed by June 30, 1998.

7. Personnel who performed the inadequate applicability reviews will be counseled on their performance. This action will be completed by March 13, 1998.

V. <u>ADDITIONAL INFORMATION</u>

A. Failed components: none.

B. Previous similar events:

LER 97-13, "Prior to 1992, Emergency Switchgear Not Seismically Qualified With Breakers Racked Out," identified a historical issue where NMP2 had racked out 4160 volt switchgear breakers in a manner which did not meet seismic qualification requirements. The cause of this error was inadequate translation of the seismic qualification program into plant procedures. LER 98-03 deals with a knowledge deficiency regarding the use of probabilistic arguments for licensing and design

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

## V. ADDITIONAL INFORMATION (cont'd)

basis criteria. These two LERs deal with different applications of seismic criteria. LER 98-03 is a historical issue that dates back to the original methodology and was not recognized as an error until the recent review of IN 97-71. As a result, the corrective actions taken in LER 97-13 would not have prevented this event.

C. Identification of components referred to in this LER:

| COMPONENT | IEEE 803 FUNCTION | IEEE 805 SYSTEM ID |
|---|---|---|
| High Pressure Core Spray System | N/A | BG |
| Reactor Coolant Recirculation System | N/A | AD |
| Residual Heat Removal System | N/A | BO |
| Reactor Water Cleanup System | N/A | CE |