NIAGARA MOHAWK

GENERATION
BUSINESS GROUP

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

November 5, 1998
NMP1L 1376

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

RE: Docket No. 50-220
LER 98-18

Gentlemen:

In accordance with 10CFR50.73(a)(2)(i)(B), we are submitting LER 98-18, "Violation of License Condition 2.C.(1) and the Power/Flow Relationship Technical Specification Due to a Degraded Valve." This report also fulfills the requirement of Technical Specification 3.1.7.g to submit a thirty day written report if the power/flow relationship of the Core Operating Limits Report is exceeded.

Very truly yours,

Robert G. Smith

Robert G. Smith
Plant Manager - NMP1

RGS/GJG/kap

xc: Mr. H. J. Miller, Regional Administrator
Mr. G. K. Hunegs, Senior Resident Inspector
Records Management

IE22

| NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION | APPROVED OMB NO. 3150-0104 EXPIRES: |
|---|---|
| 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 1 | 05000220 | 1 OF 4 |

TITLE (4)
Violation of License Condition 2.C.(1) and the Power/Flow Relationship Technical Specification Due to a Degraded Valve

| 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) |
| 10 | 06 | 98 | 98 | 018 | 00 | 11 | 05 | 98 | N/A | |
| | | | | | | | | | N/A | |

| 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) 100 | | ☐ 20.2201(b) | ☐ 20.2203(a)(2)(v) | ☒ 50.73(a)(2)(i) | ☐ 50.73(a)(2)(viii) |
| | | ☐ 20.2203(a)(1) | ☐ 20.2203(a)(3)(i) | ☐ 50.73(a)(2)(ii) | ☐ 50.73(a)(2)(x) |
| | | ☐ 20.2203(a)(2)(i) | ☐ 20.2203(a)(3)(ii) | ☐ 50.73(a)(2)(iii) | ☐ 73.71 |
| | | ☐ 20.2203(a)(2)(ii) | ☐ 20.2203(a)(4) | ☐ 50.73(a)(2)(iv) | ☐ OTHER *(Specify in Abstract below and in Text, NRC Form 366A)* |
| | | ☐ 20.2203(a)(2)(iii) | ☐ 50.36(c)(1) | ☐ 50.73(a)(2)(v) | |
| | | ☐ 20.2203(a)(2)(iv) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(vii) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER |
|---|---|
| Dave Topley - Manager Operations | (315) 349-1752 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO EPIX | | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | |

| 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 October 6, 1998, operators of Nine Mile Point Unit 1 (NMP1) determined that the actual total Control Rod Drive (CRD) System flow exceeded the indicated CRD System total flow. The reactor engineer determined that the additional 14 gpm represented approximately a 2 megawatt thermal (MWt) power increase above calculated core thermal power. This exceeded License Condition 2.C.(1) and the Technical Specification (TS) 3.1.7.d Power/Flow relationship.

The cause of this event is a degraded CRD System filter bypass valve. The degraded valve caused additional flow to the reactor that was not recorded on the CRD System total flow instrument used in the plant computer thermal power calculation.

Immediate corrective action was to limit core thermal power to 1847 MW thermal to account for the additional flow. Software changes are planned which will account for this additional flow. The valve will be replaced or repaired during the next refueling outage.

# I. DESCRIPTION OF EVENT

On October 6, 1998, as part of a procedure upgrade project, a Nine Mile Point Unit 1 (NMP1) Assistant Station Shift Supervisor (ASSS) reviewed a proposed revision to the Operating Procedure (OP) for the Control Rod Drive (CRD) System. The procedure stated that the combination of cooling water flow and hydraulic line exhaust flow should equal the total CRD System flow of approximately 65 gallons per minute (gpm). When the ASSS tried to validate that statement, he discovered a total system flow error. Using plant indications, the ASSS determined that the sum of CRD System cooling flow (45 gpm) and the CRD hydraulic exhaust flow (34 gpm) exceeded the CRD total system flow of 65 gpm. The reactor engineer calculated that the additional 14 gpm represented approximately a 2 megawatt thermal (MWt) power increase above the computer calculated core thermal power. Based upon this, License Condition 2.C.(1) and the Technical Specification (TS) 3.1.7.d Power/Flow relationship had been exceeded.

On October 8, 1998, Maintenance personnel installed portable flow measuring devices to confirm the control room indication. Based on the flow measurements, maintenance personnel determined that the CRD System filter bypass valve (28-18) leakage was 14 gpm. The bypass valve is normally closed and opened only to bypass the filters for off-normal situations. The valve was verified to be closed, therefore, the 14 gpm flow is due to valve leakage. Since it cannot be determined when the bypass valve began to degrade, the exact time frame that the thermal power limits were exceeded cannot be determined.

Since March 6, 1998, NMP1 consistently operated at a calculated core thermal power of less than or equal to 1848 MWt as a result of an event reported in LER 98-03. Prior to that time, NMP1 operated at a calculated core thermal power of 1850 MWt. Therefore, the addition of 14 gpm which equates to approximately a 2 MW thermal power would have placed NMP1 above the License Condition 2.C.(1) of 1850 MWt and above the Power/Flow relationship of TS 3.1.7.d.

As reported in LER 93-01, "Operation in Excess of 100 Percent Rated Core Thermal Power due to lack of a Procedural Requirement and Poor Man-Machine Interface," NMP1 had operated at a calculated core thermal power of 1858 MWt. Therefore, if the CRD system bypass valve was degraded as early as 1993, core thermal power could have been 1860 MWt.

# II. CAUSE OF EVENT

The root cause of this event has been determined to be degradation of the CRD System filter bypass valve. This degradation resulted in flow to the NMP1 core which bypassed the CRD System total flow input to the plant computer thermal power calculation.

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) | | | | | PAGE (3) |
|---|---|---|---|---|---|---|---|
| | | YEAR | | SEQUENTIAL NUMBER | | REVISION NUMBER | |
| Nine Mile Point Unit 1 | 05000220 | 98 | - | 18 | - | 00 | 03 OF 04 |

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

## III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications," and NMP1 TS 3.1.7.g, "Reporting Requirements," which requires a thirty-day written report be submitted if the limiting values of TS 3.1.7.d are exceeded.

The analyses for the fission product barriers (i.e., the Fuel Clad, the Reactor Pressure Vessel and the Primary Containment) at 1860 MWt is bounded by the analyses in Chapter 15, "Safety Analysis," of the Updated Final Safety Analysis Report. Both the design basis Loss of Coolant Accident (LOCA) and the design basis Containment Accident analyses assume 102 percent of rated core thermal power (i.e., 1887 MWt). These analyses demonstrate that the emergency core cooling acceptance criteria of 10CFR50.46 would be met in the event of a design basis accident at 102 percent of rated core thermal power.

Since NMP1 operated at a maximum of 100.5 percent of rated core thermal power, which was within the bounds of the design basis LOCA and Containment Accident Analyses, there were no potential consequences to the health and safety of the general public or plant personnel as a result of this event.

## IV. CORRECTIVE ACTIONS

1. NMP1 has been administratively limited to 1847 MWt to account for the additional CRD System flow.

2. The NMP1 shift check procedure has been permanently revised to require operators to observe and trend the CRD System bypass flow rate. If flow rates increase, additional derating will be imposed.

3. An independent evaluation of core thermal power heat balance parameter inputs was completed to ensure parameter validity and that system configurations similar to that reported in this LER are not present.

4. A software change will be made on the process computer to add the CRD System bypass flow rate to the heat balance calculation by November 16, 1998.

5. The CRD System filter bypass valve will be replaced or repaired during the next refueling outage.

## V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events:

LER 93-01, "Operation in Excess of 100 Percent Rated Core Thermal Power due to Lack of a Procedural Requirement and Poor Man-Machine Interface," describes an event where the Reactor Water Cleanup System flow, which is an input to the process computer, was isolated without adjusting the core thermal power calculation accordingly.

LER 95-04, "Operation in Excess of 100 Percent Rated Core Thermal Power due to Feedwater Flow Measurement Errors," describes an event where a non-conservative truncation of feedwater flow constants by the process computer and a miscalculation due to personnel error occurred.

LER 96-03, "Power to Flow/Ratio Technical Specification Violation Due to Ineffective Change Management," describes an event where the plant had operated on two occasions in April 1995 with an isolated recirculation loop that caused indicated reactor coolant flow to read higher than actual flow.

LER 98-01, Power/Flow Relationship Technical Specification Violation (Operation Above Rated Power) Due to Inadequate Management Methods," describes an event where core thermal power limits on an eight-hour average had been exceeded. An insufficiently conservative operating philosophy resulted in the operation above 100 percent power.

Due to the fact that the CRD System bypass valve degraded, the corrective actions from these previous LERs could not have prevented this event.

C. Identification of components referred to in this LER:

| COMPONENT | IEEE 803 FUNCTION | IEEE 805 SYSTEM ID |
|---|---|---|
| CRD Bypass Valve | V | AA |