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)
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TITLE (4)
Systems Outside the Design Basis Due to Incorrect Valve Weights

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)
05	25	98	98	014	00	06	24	98	N/A	05000
									N/A	05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10) 000		☐ 20.2201(b) ☐ 20.2203(a)(1) ☐ 20.2203(a)(2)(i) ☐ 20.2203(a)(2)(ii) ☐ 20.2203(a)(2)(iii) ☐ 20.2203(a)(2)(iv)	☐ 20.2203(a)(2)(v) ☐ 20.2203(a)(3)(i) ☐ 20.2203(a)(3)(ii) ☐ 20.2203(a)(4) ☐ 50.36(c)(1) ☐ 50.36(c)(2)	☒ 50.73(a)(2)(i) ☒ 50.73(a)(2)(ii) ☐ 50.73(a)(2)(iii) ☐ 50.73(a)(2)(iv) ☐ 50.73(a)(2)(v) ☐ 50.73(a)(2)(vii)	☐ 50.73(a)(2)(viii) ☐ 50.73(a)(2)(x) ☐ 73.71 ☐ OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Ray Dean, Engineering Manager - NMP2	(315) 349-4240

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 May 25, 1998, Nine Mile Point Unit 2 (NMP2) determined that differences between actual valve weights and weights shown on engineering drawings could have caused pipe stresses to exceed design allowables on four piping systems. These systems included High Pressure Core Spray (CSH), Residual Heat Removal (RHS), Reactor Core Isolation Cooling (RCIC), and Reactor Floor Drains (DFR). NMP2 was shut down in Refueling Outage 6 (RFO6) with the reactor cavity flooded and the core off loaded at the time of discovery. The systems were determined to be operable or not required for the current plant shutdown conditions with the exception of RHS Loop C. RHS Loop C was already removed from service for outage-related activities.

The root cause of this event was failure of the vendor to provide accurate valve weights during initial construction. The actual valve weights were not consistent with the vendor supplied drawings.

The valve drawings and the associated calculations were revised. Engineering Supporting Analyses were performed to determine operability. A review of other small bore valves was performed. Piping configuration changes were made such that design requirements were reestablished. The procurement process has been revised to require verification of small bore valve weights during receipt inspection.

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

I. DESCRIPTION OF EVENT

On May 25, 1998, Nine Mile Point Unit 2 (NMP2) determined that differences between actual valve weights and weights shown on engineering drawings could have caused pipe stresses to exceed design allowables on four piping systems. These systems included High Pressure Core Spray (CSH), Residual Heat Removal (RHS), Reactor Core Isolation Cooling (RCIC), and Reactor Floor Drains (DFR). NMP2 was shut down in Refueling Outage 6 (RFO6) with the reactor cavity flooded and the core off loaded at the time of discovery. The systems were determined to be operable or not required for the current plant shutdown conditions with the exception of RHS Loop C. RHS Loop C was already removed from service for outage-related activities.

In 1997, NMP2 personnel identified a discrepancy with the valve weights of small bore ASME Class 2 and 3 manual valves. The identified valves were all safety-related. The actual weights of 522 valves were determined to be higher than the weights shown on the vendor valve drawings by as much as 50 percent. The use of incorrect valve weight impacts pipe stresses, pipe support/tie-back support loads and qualification of valve accelerations. Since the valves were all manual valves, and the pipes and pipe supports/tie-back supports are passive components, the safety functions for these components consisted of maintaining structural integrity and thus the pressure boundary.

An Engineering Supporting Analysis (ESA) was performed using a sampling of various calculations that may have been impacted. Based on calculations for 400 valves that were reviewed for a variety of locations, loading conditions and configurations, and the conservatism included in the calculations, the affected valves, piping and systems were determined to meet design requirements.

The affected calculations were reviewed over a period of time to document the qualifications of the impacted piping with the new valve weights. On May 25, 1998, after the documentation of the affected calculations was completed, it was determined that of the 522 valves affected, a total of eight valves on four different systems caused the piping on those systems to not meet design requirements under normal operating and accident conditions. These affected systems included CSH, RHS, RCIC, and DFR. An additional ESA was performed which determined that the piping for six of the eight valves associated with the CSH, RCIC and DFR Systems were operable for the current plant shutdown conditions. The two remaining valves associated with RHS rendered RHS Loop C inoperable for the current plant conditions. However, RHS Loop C was already removed from service for outage-related activities.

Conservatisms used in the calculations were re-evaluated, and of the eight valves identified on May 25, 1998, three valves were determined to meet design requirements under all conditions. This included both valves associated with the CSH System and therefore, the CSH System was unaffected and capable of meeting its required functions at all times. Thus, the piping associated with only five valves did not meet design requirements for all plant conditions. These five valves are described further in the Analysis of Event section of this LER.

II. CAUSE OF EVENT

The root cause of this event was failure of the vendor to provide accurate valve weights during initial construction. The actual valve weights were not consistent with the vendor supplied drawings.

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 10CFR50.73(a)(2)(ii), "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or that resulted in the nuclear power plant being: (B) In a condition that was outside the design basis of the plant."

Technical Specification (TS) 3/4.4.8, "Structural Integrity," requires that the reactor coolant system structural integrity of ASME Code Class 1, 2, and 3 components be maintained. When the integrity of these components fails to meet the applicable requirements, the affected components must be returned to within limits or must be isolated. The piping associated with the two RHS valves (part of the reactor coolant pressure boundary) did not meet the applicable requirements since original installation due to the described deficiency, and since this condition was not recognized, the applicable TS actions were not taken. In addition, the applicable TS actions for system inoperability were not taken and other systems which were required to be operable as a result may not have been operable.

The impact of the affected valves and systems not meeting design requirements is described below:

<u>RHS Valves 2RHS*V220 and V221</u>

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

Valves 2RHS*V220 and V221 are normally closed vent valves on a three-quarter inch line on the RHS Loop C injection line. These valves are used as high point vents and also as a vent during Type C testing of the Containment Isolation Valves (CIVs). Assuming a three-quarter inch hole on the injection line during a postulated LOCA, and assuming the loss of Division I electrical power (single active failure), the RHS Loop C injection capacity would have been slightly reduced. However, such a small reduction would not have significantly affected the heat removal and core cooling function because injection flow rates used in the

III. <u>ANALYSIS OF EVENT</u> (Cont'd)

LOCA analysis are lower than current system performance test acceptance criteria. If a single active failure of the outboard CIV is assumed (i.e., post-LOCA CIV will not close), the leakage through the hole would have been confined within the RHS injection line boundary. Any leakage from the RHS line boundary (i.e., valve stem leakage) to the secondary containment would have been treated by the Standby Gas Treatment System. Therefore, changes to radiological consequences would likely have been minimal.

<u>RCIC Valve 2ICS*V225</u>

The RCIC System provides adequate core cooling in the event the reactor is isolated from its primary heat sink and feedwater flow is not available.

Valve 2ICS*V225 is normally closed and isolates a one-half inch test connection on the RCIC turbine exhaust line to the suppression pool, which is used for Type C testing of the CIV. For transients that would initiate RCIC, steam would be released from the turbine exhaust line which could lead to a RCIC isolation on area high temperature, assuming that the one-half inch connection is broken. This condition is alarmed in the control room and thus the operators would have taken the appropriate actions to place the plant in a safe condition. CSH serves as a backup to RCIC, can perform the same function as RCIC and would not have been affected by the failure in RCIC, therefore assuring the ability to place the plant in a safe condition.

The RCIC turbine exhaust line CIV is normally open. For a postulated LOCA, primary containment isolation would have been met even if the test line was to fail. The suppression pool water seal would have prevented contaminated air leakage. However, a small amount of water leakage would be expected through the opening. The consequences of such leakage are small and would likely have been bounded by current radiological analyses. Local radiation alarm and/or flooding signals would have alerted the operator to take corrective actions.

<u>DFR Valves 2DFR*V112 and V113</u>

The Reactor Floor Drains collect influent from radioactive or potentially radioactive sources and high conductivity or potentially high conductivity sources and discharge these fluids to the Radwaste System for processing.

Valves 2DFR*V112 and V113 are normally closed and isolate a three-quarter inch test connection on the floor drains leaving containment. The valves are used for Type C testing of the corresponding CIVs. This pipe connection is located in the air space above the suppression pool. The drain header is open to drywell atmosphere. For the worst case scenario, a suppression pool bypass path could have existed during a postulated LOCA. This bypass path would have resulted in an additional bypass area of approximately six percent of design. However, this additional bypass area is within the Technical Specification (TS) Limiting Condition for Operation (LCO) 3.6.2.1.b limit, which is 10 percent of the design. Additionally, past

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III. ANALYSIS OF EVENT (Cont'd)

suppression pool bypass performance tests have shown that the actual bypass area is approximately one percent of the design. Therefore, the event is within the existing limit and the containment barrier would have been assured.

During a seismic event or LOCA, the pipe stress allowables of ASME Section III Appendix F could have been exceeded and piping failures may have occurred. Although the above systems were determined to be outside the design basis due to the incorrect valve weights and resultant pipe stresses, the evaluations performed by NMP2 show that the systems would have remained functional during normal plant operation. As described above, there was adequate protection for the reactor and containment based on either redundant equipment or systems, or the minimal impact of the failures on the associated systems. Therefore, there were no adverse consequences to the health and safety of the general public or plant personnel.

IV. CORRECTIVE ACTIONS

1. The valve drawings were revised to correct the valve weights.

2. The associated calculations were revised to reflect the actual valve weights and ESAs were performed to determine operability.

3. Piping configuration changes were made such that design requirements were reestablished for the affected valves. The changes included reworking the weld contour, relocating or redesigning tie-back supports, or removing valves and installing pipe caps.

4. A review of other small bore valves supplied by this vendor was performed. Discrepancies were identified with two other valve sizes used in the plant where indicated and actual valve weights were outside an acceptable range. In one case, the valves were lighter than shown on the drawings and thus had no adverse impact. In the other case, the valves were heavier than shown on the drawings. These configurations were qualified analytically by reevaluating conservatisms used in the calculations and thus were determined to meet design requirements.

5. A requirement to verify small bore valve weights during receipt inspection has been added to the procurement process.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events: none.

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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V. ADDITIONAL INFORMATION (Cont'd)

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM ID
Residual Heat Removal System	N/A	BO
High Pressure Core Spray System	N/A	BG
Reactor Core Isolation Cooling System	N/A	BN
Reactor Floor Drain System	N/A	WK
Valves	V	BO, BG, BN, WK