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FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-001-00:on 930111,discovered that switch exceeded 100% of rated core thermal power.Caused by computer point input dropping out of computers core thermal power calculation. Deviation event repts written.W/930205 ltr.

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

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Neil S. "Buzz" Carns
Vice President
Nuclear Generation.

February 5 , 1993
NMP88327

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

RE: Docket No. 50-220
LER 93-01

Gentlemen:

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

LER 93-01 Is being submitted in accordance with 10CFR 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications;" 10CFR 50.73 (a)(2)(ii)(B), "any event or condition that resulted in the nuclear power plant being in a condition that was outside of the design basis of the plant;" and Technical Specification 3.1.7.g, which requires a 30 day written report if the power flow relationship of the Core Operating Limits Report is exceeded.

A 10CFR 50.72 report was made at 1105 hours on January 11, 1993.

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

Very truly yours,

Mr. N. S. Carns
Vice President - Nuclear Generation

NSC/JTP/lmc
Attachment

xc: Mr. Thomas T. Martin, Regional Administrator Region I
Mr. Wayne L. Schmidt, Senior Resident Inspector

Cert No P96490361

IE22 1/1

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)	DOCKET NUMBER (2)	PAGE (3)
Nine Mile Point Unit 1	0 5 0 0 0 2 2 0	1 OF 0 6

TITLE (4) Operation in Excess of 100 Percent Rated Core Thermal Power due to Lack of a Procedural Requirement and Poor Man-Machine Interface

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)
									N/A	0 5 0 0 0
0 1	1 1	9 3	9 3	— 0 0 1	— 0 0	0 2	0 5	9 3	N/A	0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 0		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	X 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)	X 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
Mr. Normal L. Rademacher, Manager Operations NMP1	AREA CODE 3 1 5 — 3 4 9 - 2 5 8 3

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)	X NO				

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

On January 11, 1993 at 1022 hours, with the mode switch in the "RUN" position, Nine Mile Point Unit 1 (NMP1) discovered that it had exceeded 100 percent of rated core thermal power (i.e., 1850 Megawatts Thermal (MWt)). Specifically, reactor power was found to be 100.5 percent of rated core thermal power (i.e., 1858 MWt). As a result, NMP1 operated in excess of the licensed maximum power level of 1850 MWt, exceeded the power-flow limiting values of the Core Operating Limits Report, and operated outside of its design basis.

The Reactor Water Cleanup (RWCU) System flow rate process computer point input dropped out of the computer's core thermal power calculation, causing NMP1 to exceed rated core thermal power. The root cause of this was the failure of multiple barriers that should have allowed operators to detect the loss of the RWCU System flow rate computer point. Contributing causes were managerial methods and procedure deficiencies.

Immediate corrective actions included reducing reactor power below 100 percent of rated core thermal power. Long term corrective actions include the requirement to perform an OD-3, "Core Thermal Power Calculation," following events or plant evolutions which may impact computer points that input into the core thermal power calculation. Additionally, training, procedure changes, and equipment repair will be performed. This event is being reported in accordance with Technical Specification 3.1.7.g as well as 10CFR 50.73.

I. DESCRIPTION OF EVENT

On January 11, 1993 at 1022 hours, with the mode switch in the "RUN" position, Nine Mile Point Unit 1 (NMP1) discovered that it had exceeded 100 percent of rated core thermal power (i.e., 1850 Megawatts Thermal (MWt)). Specifically, reactor power was found to be 100.5 percent of rated core thermal power (i.e., 1858 MWt). As a result, NMP1 operated in excess of the licensed maximum power level of 1850 MWt, exceeded the power-flow limiting values of the Core Operating Limits Report, and operated outside of its design basis.

The Reactor Water Cleanup System (RWCU) was removed from service to pre-coat #12 cleanup filter on January 9, 1993 at 12:53 hours. The system remained out of service on January 10, 1993 for troubleshooting and corrective maintenance of an unrelated item. Shortly before returning RWCU to service, reactor power was lowered to 1840 MWt. The RWCU System was returned to service on January 10, 1993 at 2125 hours. Reactor power was then increased to an indicated value of 1850 MWt at 2158 hours.

Reactor Engineering Surveillance Procedure N1-RESP-1, "Reactor Engineering Daily Thermal Limit Surveillance," was performed at the next scheduled surveillance, on January 11, 1993 at 0825 hours. The results of this surveillance indicated RWCU flow rate as a failed sensor. The process computer point for this sensor is an input to the core thermal power calculation. Thus, RWCU System flow was not considered when calculating core thermal power.

The RWCU System computer point was restored and core thermal power recalculated at 1022 hours to be 1858 MWt. This value is 100.5 percent of the rated core thermal power of 1850 MWt.

II. CAUSE OF EVENT

The RWCU System flow rate process computer point input dropped out of the computer's core thermal power calculation, causing NMP1 to exceed rated core thermal power. The root cause of this was the failure of multiple barriers that should have allowed operators to detect the loss of the RWCU System flow rate computer point. These barriers are:

1. ***Written Communication***

   There is no procedural requirement to verify that inputs into the heat balance calculation have remained functional following events which may have an impact on the core thermal power calculation.

2. ***Man-Machine Interface with the 3D Monicore Display***

   RWCU flow rate was listed on the failed sensors list of the printout attached to procedure N1-RESP-1, "Reactor Engineering Daily Thermal Limit Surveillance," performed January 10, 1993 at 07:35 hours. This was not considered abnormal

II. CAUSE OF EVENT (cont.)

since the RWCU System was isolated at this time. Prior to returning RWCU to service, the Shift Technical Advisor (STA) reviewed the heat balance that was available on the 3D Monicore (a Core Monitoring Program for the process computer) color graphics terminal. The RWCU flow rate was shown in a green box indicating that the point was "good". It was expected that a bad computer point would be displayed inside a purple box. The STA made the assumption that the computer point had been restored. It was determined that 3D color graphics will not indicate a point is deleted; instead, it displays the last "good" value sent from the Plant Process Computer.

OD-3, "Core Thermal Power Calculation," is the heat balance program available directly from the Plant Process Computer. Since 3D Monicore was installed, some operators had begun to rely on the color graphics heat balance instead of running OD-3, which would have taken the STA an additional 2-3 minutes to perform and verify. OD-3 would have properly displayed the RWCU flow rate as a failed sensor.

The uniqueness of the 3D Monicore System design relative to the heat balance visual display was not made apparent or emphasized when the system was installed.

The following factors contributed to this event:

1. ***Managerial Methods***

Response to a known and repetitive problem was untimely. RWCU filter bypass valve 35-49 had a history of failure due to problems with the remote operator linkage. The valve has been inoperable since October 31, 1989. Additionally, #11 RWCU filter has been out of service due to bad filter septums, since March 12, 1991. These deficiencies reduced RWCU System operating flexibility since only one cleanup filter (i.e., #12 cleanup filter) was available for operation. #12 filter required backwashing and precoating, so the RWCU System had to be shutdown because the filter bypass valve was inoperable in the closed position. Corrective maintenance of these deficiencies has been prioritized and scheduled in accordance with Nuclear Division directives and procedures.

2. ***Written Communication***

Operating Procedure N1-OP-03, "Reactor Cleanup System," requires the operator to establish or verify pressure of 65 to 70 psig on the cleanup pump surge tank. Pressure in the cleanup pump surge tank (downstream of the filters) was manually charged to 65-70 psig in preparation for system startup. With the system shutdown, pressure in the pre-coat portion of the system is 40-45 psig. When #12 filter was unisolated, the differential pressure caused a flow disturbance in the RWCU System's

II. CAUSE OF EVENT (cont.)

flow venturi. This flow disturbance gave indication of a failed sensor to the process computer, causing the RWCU flow rate computer point to drop out.

The last three RWCU System outages were reviewed to verify that the computer point remained functional during the RWCU System outage. In all of the cases reviewed, the computer point remained in service. The difference between this RWCU System outage and previous RWCU System outages is the action of manually charging the cleanup pump surge tank. The need for charging, which is normal, was caused by the extended duration of the January 10, 1993 outage.

The cause of the power flow relationship exceeding the limiting values of the Core Operating Limits Report was reactor power being greater than 100 percent rated power.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR 50.73 (a)(2)(i)(B), "any operation or condition prohibited by the plant's Technical Specifications," and 10CFR 50.73 (a)(2)(ii)(B), "any event or condition that resulted in the nuclear power plant being in a condition that was outside of the design basis of the plant." Further, this event is reportable per NMP1's Technical Specification Section 3.1.7.g, "Reporting Requirements," which states "if any of the limiting values identified in Specification 3.1.7(Fuel Rods)a, b, c, d(power to flow relationship), and e are exceeded, a Reportable Occurrence Report shall be submitted. If the corrective action is taken, as described, a thirty-day written report will meet the requirements of this Specification." The power-flow relationship exceeded the Technical Specification, and corrective action was taken per Technical Specification.

A 3D Monicore Predictor Log (which predicts core thermal parameters for hypothetical conditions) was run for a core thermal power of 1859 MWt, which is slightly higher than the actual reactor power. This prediction showed that core thermal limits for critical power ratio and linear heat generation rate were not exceeded for the most limiting locations in the core. Thus, the fuel clad was protected by maintaining the heat transfer mechanism of nucleate boiling and by minimizing strain on the clad. The prediction also showed that the "maximum average planar ratio" thermal limit was not exceeded.

The analyses for the fission product barriers (i.e., the Fuel Clad, the Reactor Pressure Vessel and the Primary Containment) at 1858 MWt are bounded by the accident analyses in Chapter 15, "Safety Analysis," of the updated FSAR. Both the Design Basis (DB) Loss of Coolant Accident and the DB 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 10CFR 50.46 would be met in the event of a design

III. ANALYSIS OF EVENT (cont.)

basis accident at 102 percent of rated core thermal power. Since we operated at 100.5 percent of rated core thermal power, there were no adverse consequences to the health and safety of the general public or plant personnel as a result of this event.

The duration of this event was approximately 13 hours.

IV. CORRECTIVE ACTIONS

The immediate corrective actions were to:

1. Reduce core thermal power below 1850 MWt. This also corrected the power-flow value.

2. Deviation Event Reports 1-93-0070 and 1-93-0072 were written to document the event, provide notification to management, provide for reporting requirements, initiate a root cause analysis and document corrective actions.

3. Evaluate the affect of exceeding rated core thermal power on thermal limits by running the 3D Monicore Predictor Log.

4. On-shift training of OD-3, "Core Thermal Power Calculation," was conducted for the operators of each shift.

The long term corrective actions are:

1. Require the performance of an OD-3, "Core Thermal Power Calculation," once per shift when the STA performs OD-7 option 2 for rod pattern verification. The STA shall also perform an OD-3 following events or plant evolutions which may impact computer points that input into the core thermal power calculation.

2. Perform refresher training for Control Room operators on the operation of the 3D Monicore System.

3. Procedure N1-OP-03, "Reactor Cleanup System," will be revised to ensure that RWCU System differential pressures are minimized when RWCU filters are valved into service.

4. A Lessons Learned Transmittal will be prepared to discuss and disseminate the need for complete and adequate training for modifications such as 3D Monicore.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events: none.

C. Identification of components referred to in this LER:

COMPONENT	IEEE 803 FUNCTION	IEEE 805 SYSTEM ID
Reactor Water Cleanup (RWCU) System	N/A	CE
RWCU Filter	FLT	CE
RWCU Filter Bypass Valve	V	CE
RWCU Pump Surge Tanks	TK	CE
RWCU Flow Venturi	FI	CE
3D Monicore	MON	IO