Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Entergy Nuclear Indian Point 2, LLC
P. O. Box 249
Buchanan, NY 10511

October 16, 2001

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 2001-003-00
NL-01-121

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Dear Sir:

The attached Licensee Event Report 2001-003-00 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Sincerely,

Fred Dacimo
Vice President - Operations
Indian Point 2

Attachment

cc: Mr. Hubert J. Miller
Regional Administrator - Region I
U.S. Nuclear Regulatory Commission
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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4) Instrument Calibration Error Results in Operation in Excess of Maximum Rated Thermal Power

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 NAME	DOCKET NUMBER
										05000
08	17	2001	2001	-003-	00	10	16	2001		05000

OPERATING MODE (9)	N
POWER LEVEL (10)	94

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

	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)	X	OTHER -
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard Louie, Licensing Engineer	(914) 734-5678

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).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) **(16)**

On August 17, 2001, at approximately 1442 hours, Indian Point 2 exceeded its full, steady state, licensed power level of 3071.4 MWt. Control room personnel were increasing reactor power following a power reduction to perform maintenance on a main feed pump [EIIS:SJ:P]. The power increase was stopped at 94 percent (indicated) to perform a heat balance and nuclear instrumentation [EIIS:IG] (NIS) calibration. The power increase was then resumed. At 1442, the overpower delta-temperature alarm [EIIS:IG:ALM] actuated indicating an overpower condition. As required by procedure, reactor power was reduced to clear the alarm. Another heat balance was performed indicating that reactor power was at 101.5 percent. Reactor power was further reduced to less than 100 percent. Personnel initially believed that indications of greater than 100 percent power were incorrect. Subsequent analysis concluded that reactor power had reached 102.7 percent for approximately five minutes. As a result of exceeding 102 percent of the full steady state licensed power level, this event is reportable as a violation of License Condition 2.C.(1).

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse 4-Loop Pressurized Water Reactor

EVENT IDENTIFICATION

Instrument Calibration Error Results in Operation in Excess of Maximum Rated Thermal Power

EVENT DATE

August 17, 2001

REFERENCES

Condition Reporting System Number(s): 200108052

PAST SIMILAR EVENTS

LER 2001-001-00

EVENT DESCRIPTION

On August 17, 2001, at approximately 1442 hours, Indian Point 2 exceeded its full, steady state, licensed power level of 3071.4 MWt. Reactor power had reached 102.7 percent for approximately five minutes. Control room personnel were increasing reactor power following a power reduction to perform maintenance on a main feed pump. These repairs were initiated on August 16, and necessitated a reduction in reactor power to approximately 70 percent. Upon completion of the repairs, operators initiated reactor power ascension on August 17, 2001 at approximately 1010 hours. At approximately 94 percent power (as indicated on the power range excore NIS [EIIS:IG] instruments) a heat balance was initiated per procedure SOP 15.1, "Reactor Thermal Power Calculation." This procedure requires that power be maintained steady by maintaining reactor coolant system average temperature (T AVG) and reference temperature (T REF) within one degree F during data collection and NIS calibrations. This was verified at the beginning of the heat balance, but was not maintained for the duration of the heat balance and calibrations. Approximately midway through the performance of the heat balance and NIS calibration, a turbine load perturbation occurred resulting in a decrease in turbine load, and a step rise in reactor coolant system temperature.

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EVENT DESCRIPTION (Continued)

This temperature increase resulted in higher instrument readings on the power range excore NIS instruments (as they measure neutron leakage). The power range NIS instruments were subsequently adjusted down based upon the higher than normal reading. The plant power increase was then resumed, and reactor coolant system average temperature restored to reference value. This reduced the reactor core neutron leakage and resulted in the power range excore NIS instruments being non-conservative (lower than actual plant power) by approximately 1.5 percent power. Turbine power was increased with the power range excore NIS instruments indicating approximately 98 percent power. At approximately 1424 hours, reactor power exceeded 100 percent as indicated by reactor coolant system loop differential temperature and turbine first-stage pressure (from plant computer system). Personnel initially believed that indications of greater than 100 percent power were incorrect. This overpower condition was aggravated by a subsequent step turbine load rise of approximately 20 MWe. that occurred after exceeding 100 percent power. At this time, reactor coolant system temperature decreased approximately 1.5 degrees F below reference in response to the step change in turbine power. This decreased core neutron leakage, and further aggravated the non-conservatively set NIS instruments. At 1442, the overpower delta temperature alarm actuated indicating an overpower condition. As required by procedure, power was reduced to clear the alarm. A heat balance was performed indicating that reactor power was at 101.5 percent. Reactor power was further reduced to less than 100 percent. At the time, personnel attributed the overpower event to indication problems. A condition report was initiated to document the event, and notifications made to management.

On September 14, a reactor engineering analysis prepared in response to the condition report concluded that power had reached or exceeded 102.5 percent power. Although required, a new condition report was not initiated to document this conclusion.

On September 17, operations personnel performing data collection for the monthly performance indicators noted the condition report indicating that power was at 101.5 percent. Believing this to be attributed to computer program errors, a new condition report was initiated to investigate this condition. While reviewing this condition report, the results of the reactor engineering analysis prepared on September 14 became evident. The significance level of the original condition report was escalated, and an investigation team was assembled. Final analysis based on full power delta temperatures concluded that reactor power had reached 102.7 percent for approximately five minutes.

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EVENT ANALYSIS

The root cause for this event is human performance related and is attributed to ineffective reactivity management practices during the performance of the calibration of the power range excore NIS instruments. Failure to maintain plant conditions required for the heat balance and calibration resulted in a non-conservative (lower than actual reactor power) adjustment on the excore power range NIS instruments. Procedural requirements to maintain reactor coolant system average temperature (T AVG) and reference temperature (T REF) within one degree F during data collection and NIS calibrations were not met. This resulted in operation of the reactor in excess of 100 percent.

Contributing factors were:

1. A turbine load perturbation (due to governor valve instability) occurring during the heat balance resulting in a turbine load decrease, and reactor coolant system average temperature increase. This resulted in the NIS instruments indicating a higher reactor power level than actual.
2. The operating crew treated the heat balance as a routine evolution, and no pre-task brief was performed. While a heat balance is performed each shift, they are normally performed at steady state 100 percent power. Performing the heat balance during a plant power change was not a routine heat balance and should have been identified as an error likely situation.
3. The operating crew failed to question early indications of an indicated power mismatch, question the cause of the event (perform a post-event de-brief) immediately after its occurrence, and aggressively pursue a review of the event, after the fact.

There were no structures, systems, or components that were inoperable at the start of the event that contributed to the event. All equipment functioned as designed during this event. This event did not involve any personnel injury, radiation exposure, offsite dose release, or damage to equipment important to safety.

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

EVENT SAFETY SIGNIFICANCE

This event is significant because of the unexpected increase in reactor power above the Indian Point 2 operating license basis, as specified in License Condition 2.C.(1). Based upon having exceeded 102 percent of the steady state, licensed power level of 3071.4 MWt., this event is reportable as a violation of an operating license condition.

The NIS calibration error potentially affects trip settings for instruments monitoring reactor power, which provide automatic protective actions, such as reactor trip. In this regard the consequences of exceeding the high flux, power range, reactor trip setpoint (less than or equal to 109 percent of rated power) specified in Technical Specification 2.3.1.B.(1) were evaluated. The consequences of plant operation at 102.7 percent of the rated thermal power level on the UFSAR Chapter 14 accident analyses were also evaluated. This was assessed by evaluating the probability of a plant transient occurring during the five minute period during which core power was greater than 102 percent, and by evaluating the effects on the safety analysis in the highly unlikely event that a plant transient did occur during that period.

During this event, the overpower delta T, and overtemperature delta T reactor trip logic functions were available to provide for reactor protection.

Probability of a plant transient

The probability of occurrence of a plant transient, whose impact may be affected by the elevated core power, occurring during the five minute period was calculated to be 2.1E-6 based on the most frequent initiator (total loss of main feedwater). The probability of any other initiator occurring during the five minute period would be even smaller. It should be noted that this value represents the likelihood of the transient. Although the higher power level may have represented an additional challenge to the plant, evaluations have concluded that acceptance limits would not have been exceeded. This condition would not be expected to substantially alter the response of the plant mitigation systems. Therefore, no significant change in conditional core damage probability would be expected during that five minute interval of elevated power operation.

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Consequences of a single channel reactor high flux trip setpoint of 109.6 percent

The calibration error of the power range NIS instruments resulted in the effective reactor trip setpoint for one of the four channels being 109.6 percent, which exceeds the Technical Specification limit of 109 percent. The remaining three channels were 108.9 percent, 108.2 percent, and 108.9 percent. The high flux trip is a two-out-of-four logic. Assuming a single-channel malfunction (i.e., 108.2 percent), the high flux trip setpoint would have actuated at 108.9 percent, which is within the Technical Specification limit of 109 percent. The UFSAR Chapter 14 accident analysis conservatively assumes a high flux reactor trip setpoint of 118 percent, which comprises the Technical Specification limit of 109 percent plus conservative error allowances. As such, there were no adverse consequences as a result of having a single channel high flux trip setpoint at 109.6 percent. Had the operators known that they were operating with one channel above the limit, they would have declared the channel inoperable and entered a 72 hour LCO per Technical Specification Table 3.5-2.

Consequences of a reactor power operation at 102.7 percent

The calibration error of the power range NIS instruments resulted in operation of the reactor at 102.7 percent power. The UFSAR Chapter 14 accident analyses account for a two percent calorimetric error on core power, and are typically analyzed at 102 percent power. The large break loss of coolant accident is the limiting accident for peak clad temperature and is analyzed at an uprate power of 3216 MWt. (104.7 percent). Therefore, the results pertaining to that accident remain bounding. All other pertinent Chapter 14 accident analyses were evaluated for the effect of the higher power level on departure from nucleate boiling ratio (DNBR), RCS pressure and dose consequences. A decrease in minimum DNBR could be conservatively estimated by the ratio of the power levels (102/102.7). The higher power level would have a lesser effect on RCS pressure with peak pressure not expected to change much, if at all. The dose consequences were examined to determine if there would be any increase in dose. The evaluation determined that at a reactor power slightly higher than 102 percent, significant margin would still be available to the minimum DNBR limit of 1.17, the RCS pressure limit of 2750 psia, and dose limits of 10 CFR 50.67.

Based on the above, the probability of an event occurring during the five minute period when the core power was greater than the analyzed value was determined to be very small. In addition, in the unlikely event that an accident did occur during this period, the consequences of an initial power of 102.7 percent on minimum DNBR and RCS pressure would have been within the acceptance limits. The higher initial power would have no effect on dose.

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CORRECTIVE ACTIONS

Immediate corrective action was taken to reduce indicated reactor power below 100 percent. As a result of this event, the following corrective actions have been, or will be implemented.

1. Review Station Administrative Order 442, "Reactivity Management" and revise as required to ensure actions are clearly identified upon determination of the classification for a reactivity management event. In addition to the present classifications, ensure that the significance of reaching 102 percent power is addressed. (Completed)
2. Perform a briefing/coaching of station personnel on expectations regarding questioning attitude and sensitivity to reactivity management issues. (Due Date: 10/31/01)
3. Perform a briefing/coaching of all operations crews by the Operations Manager on expectations and accountability for procedure compliance. (Due Date: 10/31/01)
4. Review this event as operating experience in initial and continuing operator training. Ensure the material covers the effects of T AVG changes on the adjustment of nuclear instruments and use of redundant indication of reactor power. (Due Date: 1/31/02)
5. Review current procedural compliance requirements against industry best practice to determine if the current standards and expectations are appropriate. (Due Date: 2/28/02)

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PREVIOUS OCCURRENCES

A review of other plant transients documented in the condition reporting system indicates no past similar overpower events. However, elements prevalent in this event were illustrated in a recent LER.

LER 2001-001: This LER reported a January 2001 turbine trip/reactivity management event during which a similar contributing factor was identified. A less than adequate procedure adherence, and questioning attitude were cited as weaknesses warranting corrective actions. The corrective actions for questioning attitude and procedure adherence taken in response to the January event were ineffective at preventing recurrence. Interim actions for the event included a debrief of all operating crews by the operations manager. This de-brief did include discussions of the identified issues of questioning attitude and procedural adherence. A review of the corrective actions for questioning attitude and procedural compliance revealed that no corrective actions were taken regarding the expectations or management monitoring and reinforcement.