Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
914 736 8003

New York Power Authority

Mr. Fred R. Dacimo
Plant Manager

August 23, 1999
IPN-99-092

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Licensee Event Report # 1999-008-00
Plant Outside Design Basis Due to an Error in an Assumption In the Main Steam Line Break Analysis

Dear Sir:

The attached Licensee Event Report (LER) 1999-008-00 is hereby submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73 (a)(2)(ii)(B).

The Authority is making no new commitments in this LER.

Very truly yours,

For

Fred R. Dacimo
Plant Manager
Indian Point 3 Nuclear Power Plant

cc: See next page

IE22

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406-1415

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700 Galleria Parkway
Atlanta, Georgia 30339-5957

U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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TITLE (4)
Plant Outside Design Basis Due to an Error in an Assumption In the Main Steam Line Break Analysis

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
07	22	1999	1999	-- 008	-- 00	08		1999	N/A	05000
									N/A	05000

OPERATING MODE (9): N

POWER LEVEL (10): 100

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)		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)
Floyd Gumble, Senior Reactor Engineer	(914) 681-6724

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 15 single-spaced typewritten lines) (16)

At approximately 1430 hours on July 22, 1999, with the plant at about 100 percent power, New York Power Authority (NYPA) made an Emergency Notification System notification to advise the Nuclear Regulatory Commission that the plant was outside design basis due to a non-conservative assumption for the main steam line break (MSLB) analysis. NYPA made the notification when informed by the Nuclear Steam Supply System (NSSS) vendor that a preliminary calculation showed that the containment design pressure would be exceeded for the MSLB unless there were changes to the design input assumptions that are part of the Indian Point 3 (IP3) licensing basis. The event was caused by failure to consider the unisolatable feedwater following the MSLB when the plant considered single failure of the main feedwater regulating valve in a 1983 analysis and considering too low a value for unisolatable feedwater in subsequent analyses. At IP3 there are 3,783 cubic feet of unisolatable feedwater piping for this case and subsequent analyses considered 800 cubic feet. The plant remained in operation because an operability determination demonstrated that, with changes to the design input assumptions, the peak calculated pressure would remain within the peak pressure currently identified on the docket. Corrective action includes performance of sampling to substantiate an assumption and performance of a safety evaluation to establish the design basis for upcoming fuel cycle based on more finalized analyses. An extent of condition review will be performed. There was no effect on the public health and safety.

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

Note: The Energy Industry identification system codes are identified within the brackets {}

DESCRIPTION OF THE EVENT

At approximately 1430 hours on July 22, 1999, with the plant at about 100 percent power, New York Power Authority (NYPA) made an Emergency Notification System notification (log 35946) to advise the Nuclear Regulatory Commission that the plant was outside design basis due to a non-conservative assumption for the main steam line break (MSLB) analysis. NYPA made the notification when informed by the Nuclear Steam Supply System (NSSS) vendor that a preliminary calculation showed that the containment [NH] design pressure would be exceeded for the MSLB unless there were changes to the design input assumptions that are part of the Indian Point 3 (IP3) licensing basis. The plant remained in operation because an operability determination demonstrated that, with changes to the design input assumptions consistent with current plant conditions, the peak calculated pressure would remain within the peak pressure currently identified on the docket. Deviation Event Report (DER) 99-1485 documented the event.

This event was identified due to a R.E.Ginna plant report, on March 24, 1999, of a condition outside the plant design bases due to two non-conservative assumptions in the NSSS vendor MSLB analysis. Ginna reported that modeling errors were associated with the MSLB. The model did not properly consider the volume of feedwater between the main feedwater pump [P] discharge valve [V] (MFPDV) and the faulted steam generator [SG] (SG) for the case where there is a failure of the main feedwater regulating valve (MFRV) to the faulted SG. The model also isolated feedwater in a time frame that did not consider isolation valve closure times. The NSSS vendor opened a problem identification tracking number to address the Ginna event. This was closed in April 1999 prior to identification of this event at IP3. NYPA, based on the Ginna part 21 report, initiated DER 99-853 on April 30, 1999 to track the evaluation of the significance to IP3.

The NYPA evaluation determined that our existing MSLB analysis adequately considers the isolation time for feedwater but does not properly address the unisolatable volume of feedwater. NYPA calculated the volume of feedwater piping between the MFPDV to the SG as 3,783 cubic feet. This information was supplied to the NSSS vendor around June 1, 1999 and the NSSS vendor advised that conservatisms in the computer code were expected to account for the increased volume of water. On July 22, 1999, the NSSS vendor advised NYPA that the peak calculated pressure from the MSLB would exceed our design bases unless changes were made to our licensing

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basis assumptions. At that time, they advised that our existing peak calculated pressure would not be exceeded if credit was taken for margin in our current licensing basis. The assumptions changed were the 0 parts per million boron (ppm) assumed in the high head safety injection system [BQ] (HHSI) (the FSAR says unborated water) and the 1.3 percent shutdown margin (the FSAR and Technical Specification 3.10.1.1 say 1.3 percent). The NSSS vendor changed these to 2300 ppm boron and 2.6 percent shutdown margin. The NSSS vendor subsequently revised the evaluation to change the shutdown margin to 2.9 percent and the 2300 ppm boron assumption back to 0 ppm from the HHSI pumps on the non boron injection tank [TK] header (BIT) and from the BIT on the BIT header. The plant continues to operate based on the operability determination. The increase in shutdown margin is applicable only during power operation because the feedwater heaters do not have steam and are a much smaller contributor to the peak pressure. NYPA noted that there are conservatisms in the calculation. The calculation assumed: MFPDV shut in 120 seconds (they have consistently shut in less than 54 seconds when tested); there is full feedwater flow until the MBFPDV shuts (the valves would start to decrease flow prior to that time); and, river water of 95 degrees Fahrenheit (river water has not exceeded 90 degrees Fahrenheit from 1991 until the present).

The original plant evaluation for the MSLB included failures of the main steam non-return check valve and one train of safety injection and containment heat removal. WCAP-8822, Mass and Energy Releases Following a Steamline Rupture," September 1976, evaluated the MSLB but did not postulate failure of the MFRV. The unisolatable feedwater was based on a generic assumption of 800 cubic feet from the MFRV to the SG for four loop plants like IP3. In response to Bulletin 80-04, NYPA submitted, on March 31, 1983, the results of a MSLB analysis, performed by NYPA, that considered the effects of a MFRV failure. The analysis was reviewed and it could not be determined if the unisolatable feedwater flashing phenomena was considered. NYPA concluded that this phenomena was not considered. The NSSS vendor has continued to use the value of 800 cubic feet from WCAP 8822 in subsequent MSLB analyses even though the MFRV was a changed failure. Also, NYPA failed to identify the 800 cubic feet as an erroneous assumption during routine parameter reviews.

EXTENT OF CONDITION

WCAP-8822 represents the original methodology and data base to produce mass and energy release transients and is the source of the

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original assumption of 800 cubic feet of unisolatable feedwater. NYPA will review WCAP-8822 to determine whether there are any additional assumptions that may not be conservative with respect to IP3.

CAUSE OF EVENT

The cause of the error in the 1983 analysis as well as the subsequent failure of the NSSS vendor and NYPA to identify the erroneous assumption is apparently human error. Due to the passage of time, the factors causing the human error could not be established.

CORRECTIVE ACTIONS

The following corrective actions have been completed or will be performed under the corrective action program and will address the cause of the event:

- Prepare a Nuclear Safety Evaluation (NSE) to adopt the final NSSS vendor calculation for the upcoming fuel cycle (scheduled to start outage September 10, 1999) prior to startup from refueling. This currently would require a Technical Specification and FSAR change as well as a procedure for sampling.

- A sample was taken from the HHSI lines upstream of the pumps and at the BIT tank that verfied the boron assumption.

- Review the current program for reviewing NSSS vendor analytical assumptions to determine whether it is currently adequate to identify erroneous assumptions of the type reported and schedule corrective action if needed.

- Review WCAP-8822 to identify if there are any non-conservative assumptions with respect to IP3 and initiate corrective action if there are.

ANALYSIS OF EVENT

The event is reportable under 10 CFR 50.73(a)(2)(ii)(B). The licensee shall report any operation or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. The peak calculated pressure for the MSLB exceeds the design pressure of the containment when calculated

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using licensing basis assumptions of 1.3 percent shutdown margin and no boron in the HHSI lines. The plant is considered outside the design basis since original plant operation because the effects of the worst case single failure have not been identified until this event. The plant remains outside design basis until the NSE is completed. Plant operations continue based on an operability determination.

A review was performed of Licensee Event Reports (LERs) for the past three years to determine where the plant was outside the design basis due to original plant design. LER 97-006 reported that operation of the Refueling Water Storage Tank (RWST) purification loop was outside the design basis since original provisions to isolate the non-safety purification loop did not meet design criteria. LER 99-006 reported that routing of the component cooling water inside the missile shield wall was outside the design basis since the design criteria required routing outside the shield wall. The corrective actions for the events are not related to the event reported in this report.

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public

There were no actual safety consequences for the event. There was no MSLB that could challenge containment. An operability determination demonstrated that the peak calculated containment pressure remains within the plant design basis for the current cycle. The shutdown margins exceeded 2.9 percent for some prior cycles (cycle 9, 6 and 5) but not for others (cycle 8 and 7). However, for cycle 8 and earlier, the boron injection tank (BIT) was in service and provided a large input of negative reactivity subsequent to SI injection. Although analyses are not being reperformed for these cores, engineering judgement suggests that the BIT would compensate for the differences in shutdown margin. Other margin is also available for compensation (e.g., analyses are performed with the most reactive rod stuck).

The event was assessed using the guidance of NEI 99-02 (draft Rev. B), dated May 1999, "Regulatory Assessment Performance Indicator Guideline" as a potential safety system functional failure. It was determined that no safety system functional failure occurred since the containment could have performed its design function.