Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
914-736-8000

New York Power Authority

August 11, 1995
IPN-95-085

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 Number 95-014-00
"Low Pressure Operation Placed the Plant Outside Design Basis Due To Inadequate Procedures and Improper Document Use"

Dear Sir:

The Power Authority is submitting the attached Licensee Event Report (LER) 95-014-00, as required by 10CFR50.73. This event is the type defined in 10CFR50.73(a)(2)(ii)(B).

Also, attached are the commitments made by the Power Authority in this letter.

The Power Authority recognizes the implications of the human performance errors during this event. We will perform a root cause evaluation for this event that will encompass errors from similar events to enhance the human performance in plant operation. If this root cause significantly changes the LER causes or corrective actions, we will supplement the LER within thirty days of completion.

If there are any questions, please contact Mr. K. Peters of my staff at (914) 736-8029.

Very truly yours,

L. M. Hill
Site Executive Officer
Indian Point 3 Nuclear Power Plant

Attachments
cc: See next page

IE22

cc: Mr. Thomas T. Martin
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
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U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

NRC FORM 366 (5-92)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point Unit 3	05000286	1 OF 10

TITLE (4) Low Pressure Operation Placed The Plant Outside Design Basis Due To Inadequate Procedures And Improper Document Use

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
07	10	95	95	-- 014 --	00	08	11	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	052	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iv)	✓ 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 (Include Area Code)
Floyd Gumble, Senior Reactor Engineer I	(914)681-6724

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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)**

On July 10, 1995, at 1025 hours, with the reactor at 52 percent power, management had operators reduce the Reactor Coolant System (RCS) pressure from normal operating pressure (2235±30 psig) to about 1950 psig in an attempt to reseat leaking relief valves in the RCS. The plant subsequently operated at reduced pressure for about 51 hours during repeated attempts to reseat the leaking valves and, during that period, increased power to 59 percent. On July 12, Corporate Reactor Engineering decided that the plant had been placed in an unanalyzed condition by operating at power with RCS pressure below 2205 psig. Management decided to return the RCS pressure to normal, commence a power reduction and to make a 1-hour notification to the NRC. Later that day, management decided to bring the plant to hot shutdown to reseat the valve and perform other work. Subsequently, Engineering had Westinghouse perform an analysis that showed the plant would not have exceeded design basis limits if any postulated design basis transient or accident had occurred while it was in the reduced pressure condition for the operating power levels. Therefore, the event had no significant effect on the health and safety of the public.

The cause of this event is a combination of inadequate procedures, inadequate procedure adherence, misapplication of the plant Technical Specifications, incomplete communications and incomplete understanding and use of documents representing the plant's design basis. Corrective actions include procedural reviews and revisions, compilation of design basis assumptions for training, a Technical Specification amendment, lessons learned critiques and training. Additionally, the Independent Safety Engineering Group is augmenting its oversight role on plant operations, as an interim measure.

DESCRIPTION OF EVENT

On July 10, 1995, at 0428 hours, with the reactor at 89 percent power, during performance of surveillance test 3PT-V11, "Overtemperature Delta-T and Overpower Delta-T Calibration," a turbine runback was inadvertently initiated. The runback brought the reactor from 89 percent to about 49 percent power. Also, during the runback, there was a Reactor Coolant System (RCS)(AB) pressure increase from normal operating pressure (2235±30 psig) to about 2350 psig, which resulted in the opening of the pressurizer power-operated relief valves (PORVs) at about 2335 psig for a few seconds. The safety relief valves(RV) setpoint is 2485 psig, within 1 percent, and therefore the runback did not challenge them. As a result of the runback, the common tailpipe temperature indicator for PORVs and tailpipe temperature indicators for the three pressurizer safety relief valves had increased above 200 degrees F, which shows a reactor coolant leak through the valves. The pressurizer safety relief valves tailpipe temperature increased above the normal temperature range but below the alarm setpoint of 250 degrees F (250 degrees F is the point that requires operator action under Alarm Response Procedure ARP-3), and one safety relief valve tailpipe temperature returned to normal. However, the PORVs tailpipe temperature increased to about 264 degrees F during the runback and then decreased below 250 degrees F. The PORV's tailpipe temperature was above 250 degrees F for about 6 minutes.

On July 10, during the beginning of the day shift, management made a decision to reduce RCS pressure temporarily to no less than 1900 psig to reseat the relief valves, with the intention of meeting the safety relief valve manufacturer's recommendation. They made the decision to stop the leak, preclude increased leakage and prevent damage to the valves from steam cutting. Management believed justification existed to support the decision not to bring the plant to hot shutdown. This was based on their interpretation of Technical Specification (TS) Figure 2.1-1 and its basis, allowing operation within the temperature and power regime as a function of RCS pressure. Also, Plant Operating Procedure POP-2.1, "Operations at Power," and Alarm Response Procedure ARP-3, "Reactor Coolant System," did not place any restrictions on reactor power during RCS pressure reduction. There is no technical specification in the Indian Point Unit 3 license that establishes minimum RCS pressure for normal operation.

At 1025, the operators began reducing RCS pressure from normal operating pressure (2235±30 psig)to about 1950 psig and stayed at that pressure for approximately one hour in an attempt to reseat the relief valves. The reduction in pressure caused the leaking valves to reseat, as indicated by a reduction in tailpipe temperatures. The operators then began increasing RCS pressure, but at approximately 2100 psig, they observed the relief valves leaking again. Management had the operators reduce RCS pressure to 1950 psig for an extended period of soaking (about 18 hours).

On July 11, at 0930 Operations sought additional information on power operation with reduced RCS pressure from Licensing. About the same time the Operations Manager asked the System Engineering Manager to look at the appropriateness of low pressure operations. At about 1030 Licensing contacted the Corporate Reactor Engineering Department (the department responsible for the Indian Point 3 transient analyses) to ask if short-term operations according to ARP-3 (i.e., pressure reduction to 1900 psig, an 8-hour soak and return to operating pressure at 50 psig increments) was consistent with the DNBR assumptions of the plant design basis transient and accident analysis as reflected in the FSAR. Reactor Engineering believed that this was acceptable for short-term operation for the purposes of valve reseating, based on TS Figure 2.1-1 and the low probability of a transient occurring during this period. Reactor Engineering did not specify additional guidance for the duration allowed in low pressure operation because their review was addressing the operation allowed by the procedure that specified an 8-hour soak time for reseating the relief valve.

The Corporate Reactor Engineering response was based on a technical review of Technical Specifications Figure 2.1-1, which established the temperature/pressure/power limitations beyond which safety limits are exceeded. Reactor Engineering reasoned that the probability of a design basis transient occurring at reduced pressure would be extremely low if the duration of low pressure operation was for the short-term.

On July 11, at 1200 hours, after the extended soak, Operations commenced incremental returning of pressure. At 1518 hours, the operators observed valve leakage at about 2050 psig. At this point, they reduced RCS pressure to 2000 psig, reseating the valves, and held there. That afternoon, management considered alternate corrective actions including long-term operation at reduced pressure, operating at normal pressure with valve leakage, and shutting down to remove and refurbish the safety valves. Since other plants have operated at reduced pressure for long-term, the approach was to establish the needed analysis and documentation for operating long-term.

On July 12, early in the day shift, the System Engineering Department initiated a request to the on-site Westinghouse representative asking what was needed for continued operation at low pressure. At 0840 hours, with the leaking valve seated and the RCS still at 2000 psig, management believed that plant conditions were acceptable for power ascension. They had the operators increase reactor power from 52 percent (at the rate of about 3 percent/hour) according to normal power ascension procedures and achieved 59 percent at 1030 hours and held there awaiting the return to service of certain secondary side equipment.

At approximately 0930 hours, the Operations Department contacted the Licensing Department to question the acceptance of continued operation at reduced pressure. Licensing with Operations contacted Corporate Reactor Engineering to ask about long-term operation at reduced RCS pressure. Reactor Engineering responded that a longer period of operation would require an evaluation by Westinghouse. Pending contacting Westinghouse, Reactor Engineering believed the current condition was acceptable based on the Technical Specification and the low probability of a transient. Immediately thereafter, Reactor Engineering telephoned Westinghouse. Subsequent telephone conversations with Westinghouse confirmed that operation in this mode was not justifiable without an analysis. In other words, any transient initiated at reduced RCS pressure would put the plant in an unanalyzed condition that might exceed design basis requirements. The Standard Technical Specifications allows deviations from normal RCS pressure during power ramping of greater than 5 percent rated thermal power per minute, power steps greater than 10 percent, and for up to two hours or shutdown within the next six hours. The allowance for the deviations includes the basis that operation for a significant period outside the parameter increases the likelihood of fuel damage resulting from a design basis event occurring at reduced pressure.

Corporate Reactor Engineering contacted Indian Point Unit 3 plant and corporate management at approximately 1100 hours to inform them the plant was in an unanalyzed condition. At 1115 hours, the Shift Manager directed the operators to return the RCS pressure to normal operating pressure and commence a power reduction. Restoration of normal pressure was at 1323 hours. By this time, the RCS pressure had been continuously below 2235 psig for a total period of about 51 hours.

At about 1200 hours, Corporate Reactor Engineering prepared DER 95-1632 to document this event. Management decided, based on the information available, they could not determine the extent of the effect on plant safety. Based on this decision, at 1310 hours, Operations made a one-hour notification to the NRC under 10CFR50.72(b)(ii)(A), on the basis that the plant was in an unanalyzed condition that potentially affected plant safety.

At 1300 hours, the Chief Nuclear Officer met with corporate and plant management at the site to evaluate the circumstances and decide upon the appropriate actions to take. At 1910 hours, plant management directed operations to take the plant to hot shutdown to reseat the valves and perform other work. Operators commenced the shutdown and the plant was off-line at 0118 hours and the reactor was subcritical at 0145 hours on July 13. Pressure was reduced according to the safety relief valve manufacturer's recommendation and the safety valves leakage stopped.

Subsequently, the plant returned to normal operating pressure range and the safety valves are not leaking. One PORV block valve is shut isolating a PORV's leakage; the other PORV is not leaking.

Corporate Reactor Engineering evaluated the safety significance of this event that includes a Westinghouse analysis bounding the actual operating parameters from July 10 through July 12. On August 8, 1995, Westinghouse submitted a Justification of Past Operation (JPO) to the Authority that shows design basis limits were not exceeded for the postulated design basis transients and accidents while in the reduced pressure condition for the operating power level up to 60%. Westinghouse is notifying all Westinghouse plants of this event via an Infogram.

CAUSE OF THE EVENT

The Power Authority has determined that this event was due to a combination of the following causes:

Inadequate Procedures: The procedure POP- 2.1 did not specify pressure limitations for normal operation based on the design basis assumptions used for the plant accident analyses. Procedure ARP-3 did not have a valid engineering basis to specify reducing pressure for reseating safety valves while in the power operation condition.

Inadequate Procedure Adherence: Operations continued prolonged operation beyond the period specified in ARP-3. Using the ARP-3 as guidance to support the decision for prolonged operation of the plant at reduced pressure is an inappropriate application of the procedure. Also, during the previous two procedure revisions of ARP-3, Operations added a pressure reduction value to the step (which always had allowed pressure reduction without a specified value to reseat the valves) and did not adequately answer the safety applicability screen according to administrative procedure AP-3, "Procedure Preparation, Review and Approval." Therefore, they did not perform a 50.59 evaluation.

Misapplication of the Technical Specifications: Figure 2.1-1 and its basis "show the loci of points of thermal power, RCS pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid." Furthermore, the basis of Section 2.3 states, "the overtemperature delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 3.5 seconds) and (2) pressure is within the range between high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors, is always below the core safety limit as shown on Figure 2.1-1." IP3 staff incorrectly interpreted the words as justification for operation below normal RCS pressure. Since there is no technical specification defining minimum RCS pressure for normal operation, there was no specification that contradicted this interpretation.

In addition, because managers knew of other utilities that operated at reduced RCS pressure, this supported the mindset that similar relaxation of RCS pressure was acceptable at Indian Point 3.

Incomplete Communications: Plant Management did not solicit Corporate Reactor Engineering and Licensing early in the decision process before reducing pressure during power operation. Corporate Reactor Engineering did not clearly define limitations for low pressure operation. Operations, Licensing and Corporate Reactor Engineering were not fully effective in communicating the current operating parameters and specifying limitations on periods of low pressure operation when discussing reseating of the safety valves.

Incomplete Understanding and Use of Documents Representing the Plant's Design Basis: The decision to operate at power with a reduced RCS pressure was made without a complete understanding of all the implications of operation in this manner. The FSAR and Westinghouse were not consulted while making the decision to reduce pressure. Knowledge of, and training on, analytical assumptions for transient accident analyses, safety limits and margins are deficient.

The contributors to this lack of understanding are the causes listed above and the Authority concludes that operation in an unusual or uncertain condition, except during an emergency according to 10CFR50.54(x), must involve greater cognizance over a wider range of supporting staff and requires evaluation in advance using applicable documents.

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

CORRECTIVE ACTIONS

Indian Point 3 Performance Enhancement Review Committee held a meeting (on July 20) to evaluate the event, identify inappropriate actions and identify corrective actions. Department Managers conducted a Lessons-Learned critique (on July 24).

The Chief Nuclear Officer has directed, for an interim period, the Independent Safety Engineering Group representatives to evaluate daily the current equipment status and operating parameters of Indian Point Unit 3 and to review plans for change to any of these parameters over the following 24 hours. These individuals will be responsible for notifying senior site and corporate managers when there is reason to believe that current or future operations may be encroaching upon the limits of normal operating patterns.

The Operations Department changed procedure ARP-3 to specify maintaining RCS pressure and restoring it above 2205 psig within 2 hours or shutdown to hot shutdown within the next 6 hours.

The Operations Manager issued a Standing Order 95-05, "Design Basis Compliance," to require operation of safety-related systems and components within proceduralized operating ranges and the requirement for a formal review if not able to maintain the range.

The Operations Department will revise Administrative Procedure AP-21, "Conduct of Operations," by August 30, 1995, to incorporate the Standing Order 95-05.

The Operations Department will revise Plant Operating Procedures to specify the normal operating ranges for key operating parameters. Operations will complete the revisions by September 15, 1995.

The Configuration Information Department will evaluate the use of the safety application screens for procedure revisions and revise the associated procedure AP-3 to strengthen the screening process by September 30, 1995.

The Training Department is training licensed operators and Shift Technical Advisors during their requalification training cycle on lessons learned (including affirming procedure adherence) from this event by September 30, 1995.

The Power Authority recognizes the implications of the human performance errors during this event. We will perform a root cause evaluation for this event that will encompass errors from similar events to enhance the human performance in plant operation. We will complete the root cause by September 30, 1995. If this root cause significantly changes the LER causes or corrective actions, we will supplement the LER within thirty days of completion.

Corporate Reactor Engineering is preparing a compilation of plant conditions and equipment operability assumed in the plant transient and accident analyses. They will complete this compilation by October 27, 1995, and submit to Training and Design Engineering.

IP3 Design Engineering will review selected Operations procedures (Alarm Response Procedures (ARPs), Plant Operating Procedures (POPs) and Off-Normal Operating Procedures (ONOPs)) to identify other potential operating conditions that may require further evaluation. During the review, they will forward findings to Operations for procedure revisions and complete the review by December 15, 1995.

The Training Department will train licensed operators and Shift Technical Advisors on the plant conditions and equipment operability assumed in the plant transient analyses by December 15, 1995.

The Training Department will train Senior Managers of Technical Groups, supervisors of system engineers and reactor engineers supporting IP3 on the operating limits of the plant. Training will complete this training by December 29, 1995.

The Licensing Department will prepare and submit a Technical Specification Amendment to define operating pressure limits by December 29, 1995.

The Operations Department will complete procedure revisions by January 31, 1996, based on the Engineering's review and findings on ARPs, POPs and ONOPs.

ANALYSIS OF THE EVENT

This event is reportable under 10CFR50.73(a)(2)(ii)(B) because Indian Point Unit 3 was "in a condition outside the design basis of the plant." The Power Authority made the one-hour notification under 10CFR50.72(b)(ii)(A), for the plant being in an unanalyzed condition based on the decision that it may compromise plant safety. However, a JPO prepared by Westinghouse has shown that, based on an accident-by-accident evaluation or analysis for actual plant pressure and power operating conditions, the plant would not have exceeded design basis limits for the postulated design basis transients and accidents. Nonetheless, the plant was placed in an unanalyzed condition, outside design basis, by operating steady state for about 51 hours with RCS pressure below 2205 psig, and therefore this event is reportable under 10CFR50.73(a)(2)(ii)(B).

The following are other LERs similar to this event where actions were taken without adequate consideration of requirements: LER 93-011, 93-015, 93-019, 93-24, 94-010 and 95-001.

SAFETY SIGNIFICANCE

This event had no significant effect on the health and safety of the public. The Westinghouse analysis/JPO, which was received on August 8, 1995, shows that, no design basis accident or transient would have placed the plant outside design basis limits, were it to have happened any time when the plant was at or below 60 percent power with the RCS pressure as low as 1900 psig.

List of Commitments

Number	Commitment	Due
IPN-95-085-01	The Operations Department will revise Administrative Procedure AP-21, "Conduct of Operations," to incorporate the Standing Order 95-05.	August 30, 1995
IPN-95-085-02	The Operations Department will revise Plant Operating Procedures to specify the normal operating ranges for key operating parameters.	September 15, 1995.
IPN-95-085-03	The Training Department is training licensed operators and Shift Technical Advisors during their requalification training cycle on lessons learned (including affirming procedure adherence) from this event.	September 30, 1995
IPN-95-085-04	The Configuration Information Department will evaluate the use of the safety application screens for procedure revisions and revise the associated procedure AP-3 to strengthen the screening process.	September 30, 1995
IPN-95-085-05	The Power Authority recognizes the implications of the human performance errors during this event. We will perform a root cause evaluation for this event that will encompass errors from similar events to enhance the human performance in plant operation. If this root cause significantly changes the LER causes or corrective actions, we will supplement the LER within thirty days of completion.	September 30, 1995
IPN-95-085-06	Corporate Reactor Engineering is preparing a compilation of plant conditions and equipment operability assumed in the plant transient analyses. They will submit the compilation to Training and Design Engineering.	October 27, 1995
IPN-95-085-07	The IP3 Design Engineering will review selected Operations procedures (Alarm Response Procedures (ARPs), Plant Operating Procedures (POPs) and Off-Normal Operating Procedures (ONOPs)) to identify other potential operating conditions that may require further evaluation. During the review, they will forwarded findings to Operations for procedure revisions.	December 15, 1995
IPN-95-085-08	The Training Department will train licensed operators and Shift Technical Advisors on the plant conditions and equipment operability assumed in the plant transient analyses.	December 15, 1995

Number	Commitment	Due
IPN-95-085-09	The Training Department will train Senior Managers of Technical Groups, supervisors of system engineers and reactor engineers supporting IP3 on the operating limits of the plant.	December 29, 1995
IPN-95-085-10	The Licensing Department will prepare and submit a Technical Specification Amendment to define operating pressure limits.	December 29, 1995
IPN-95-085-11	The Operations Department will revise procedures based on the Engineering's review and findings on ARPs, POPs and ONOPs.	January 31, 1996