Indian Point 3
Nuclear Power Plant
P.O. Box 215
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NewYorkPower Authority

L. M. Hill
Site Executive Officer

September 11, 1995
IPN-95-094

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 # 95-016-00
Total System Leakage Greater Than Technical Specification Limit and Design Basis Limit For Control Room Habitability due to Technical Specification and Procedure Inadequacies

Dear Sir:

The attached Licensee Event Report (LER) 95-016-00 is hereby submitted as required by 10CFR50.73. These events are of the type defined in 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(B). Also, attached are the commitments made by the Authority in this LER.

Very truly yours,

James Zoch for

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

Attachment

cc: See next page

JE22 1/1

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

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Atlanta, Georgia 30339-5957

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

Attachment I
List of Commitments

Number	Commitment	Due
IPN-95-094-01	A proposed Technical Specification change will be submitted to incorporate any necessary clarifications with regard to which external post accident systems need to be monitored.	4/30/96
IPN-95-094-02	A proposed Technical Specification change will be submitted to reduce the maximum allowable leakage from external post accident systems to a value that will ensure post accident control room habitability.	4/30/96
IPN-95-094-03	Procedure DCM-2 will be revised such that it will require that when a calculation establishes a new operating limit, the new operating limit is identified and affected procedures and documentation that would require revision because of this new operating limit are identified.	3/29/96
IPN-95-094-04	The Authority will validate at least one system versus its Design Basis Document. This validation will determine the consistency between plant operating, surveillance, and maintenance activities and procedures and the as-built design basis information in the Design Basis Documents.	12/31/96

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 8

TITLE (4) Total System Leakage Greater Than Technical Specification Limit and Design Basis Limit For Control Room Habitability due to Technical Specification and Procedure Inadequacies

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
08	11	1995	95	016	00	09	11	95	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	100

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

	20.402(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)
Sharon P. Murray	(914) 736-8010

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 August 11, 1995, a Deviation Event Report (DER) was written to identify two periods (April/May 1990 and October/November 1992) when the unit was at power operation and external post accident system leakage rates exceeded the design basis limit for control room habitability. During the April/May 1990 period, the leakage rate also exceeded the technical specification limit of 2 gph. LER 92-005-00 reported an additional event (in April 1992) in which the leakage rate exceeded the technical specification limit. Due to procedure inadequacies and technical specification inadequacies, these non-compliances were not identified at the time of the events. The increased leakage in these events was due to: 1) a Safety Injection (SI) Pump seal (April 1992 event); 2) a Residual Heat Removal (RHR) Pump seal (April/May 1990 event); and 3) an RHR system leak (October/November 1992 event). Corrective actions for these events include Technical Specification changes and procedure revisions.

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

DESCRIPTION OF THE EVENT

On April 20, 1992, with the unit at cold shutdown (reactor coolant system (RCS) temperature 170 degrees F at atmospheric pressure), tabulation of total leakage for external post accident systems was calculated to be 2.24 gph which exceeded the Technical Specification limit of 2 gph. (This was reported in Licensee Event Report (LER) 92-005-00.)

The calculation of system leakage was directed by surveillance procedure 3PT-C1, Total Leakage Rate Monitoring Tabulation. The program consists of a review of surveillance tests associated with the components of the external post accident system.

Safety Injection Pump functional test 3PT-M16 was performed on April 14, 1992, while the unit was at 100% power. After the test was reviewed and the pump determined to be operable, the data was then incorporated into 3PT-C1, Revision 6, for total leak rate calculation. No leakage of the seal was experienced prior to the surveillance test. The results of the 3PT-C1 tabulation on April 20, 1992, indicated a leak rate of 2.24 gph which exceeded the Technical Specification limit by 0.24 gph. The sources of leakage included 0.65 gph from various sources and 1.59 gph from 31 safety injection pump (SI)(Pacific Pumps Model JTCH)(BQ)(SEAL)(P025) inboard seal.

On April 20, 1992, when the 3PT-C1 calculation was performed, the plant was in the cold shutdown condition for a refueling outage. As stated, a 30 day report was made as required by 10CFR50.73(a)(2)(i)(B)(LER 92-005-00).

Subsequent to the submittal of LER 92-005-00, as part of the design basis reconstitution effort to determine the operability requirements of the Primary Auxiliary Building (PAB) Ventilation and Filtration System, the Power Authority determined a control room habitability calculation included an assumption that was not consistent with the plant technical specifications. The assumption, made by contractor personnel, was that post accident external recirculation system leakage would be 0.7 gph, the actual leakage at the time the calculation was performed. At the time the calculation was performed, it appears as though it was not recognized that the technical specifications allowed post accident external recirculation system leakage to be 2 gph. The original calculation (assuming 0.7 gph) concluded that post accident operation of the PAB Ventilation and Filtration System was not required for control room habitability, but the limiting leak rate was not determined and nothing further was done with the 0.7 gph value.

Additionally, the control room habitability calculation was not identified as establishing a new operating limit for external post accident system leakage.

In order to resolve the above discrepancy, a new control room habitability calculation was performed and the Power Authority determined that in order to ensure post accident control room habitability without the use of the PAB ventilation system, the external recirculation system leakage should be limited to 1.34 gph. This calculation was approved on June 29, 1995.

Past leakage rates of the external recirculation system were reviewed to determine whether the plant was outside the design basis in the past. On August 11, 1995, two instances, in addition to the instance in April 1992, in which the external recirculation system leakage exceeded the recalculated design basis limit for control room habitability (1.34 gph) were documented in DER 95-1851. These events are described below.

During the period from April 11, 1990, through May 5, 1990, while the plant was at power operation, the external recirculation system leakage exceeded the recalculated design basis limit for control room habitability (1.34 gph) and exceeded the Technical Specification limit of 2 gph. During this period, the leakage was as high as 2.62 gph due primarily to a seal leak on the 31 RHR pump (BP)(SEAL). From April 11, 1990, through May 5, 1990, the plant remained at 100% power with the exception of the period from April 23, 1990, through April 25, 1990, when reactor power was between approximately 75% and 100%. During this event, the leakages from external post accident systems were summed but not compared to the technical specification limit. The failure to meet the Technical Specification limit of 2 gph during this period should have been reported in LER 92-005-00.

During the period from October 16, 1992, through November 2, 1992, while the plant was between approximately 70% and 100% power, the external recirculation system leakage exceeded the recalculated design basis limit for control room habitability (1.34 gph). During this period, the leakage was 1.47 gph (less than the technical specification limit of 2 gph) due primarily to RHR system leakage (BP).

CAUSE OF THE EVENT

In each of the three events described in this LER, there were mechanical failures which caused the leakage to exceed the leakage limits. These mechanical failures were corrected.

In the April 1992 event (reported in LER 92-005-00), the cause of the delay in recognizing that the leakage exceeded the Technical Specification limit from April 14, 1992, through April 20, 1992, was inadequate surveillance procedures at that time. The surveillance procedures that were used to measure external post accident system leakage were inadequate in that they failed to include total external post accident system leakage as a component of the operability determination of the procedure.

In the April/May 1990 event, the cause of the failure to identify that external post accident system leakage was in excess of the technical specification limit (2 gph) was that Technical Specification 4.4.I had not been interpreted as applying to all external post accident systems. The specific wording in Technical Specification 4.4.I is as follows: "The maximum allowable leakage from the Residual Heat Removal System components located outside of the containment shall not exceed two gallons per hour." During the time period in April/May 1990, when external post accident system leakage was greater than 2 gph, the leakage attributed to the RHR system was below 2 gph. During this period, RHR leakage reached 1.92 gph. Since that time, a Technical Specification Interpretation (TSI) has been written that states that the 2 gph limit is the maximum allowable leakage from RHR components and Safety Injection System components located outside containment and used during the recirculation phase of a design basis accident.

In the October/November 1992 event, as well as in the two events described above, the cause of the failure to use the appropriate design basis limit as the external post accident system leakage limit was that the original control room habitability calculation used the actual external post accident system leakage of 0.7 gph instead of the technical specification allowed limit of 2 gph and the person performing the calculation apparently failed to recognize that the calculation established a new operating limit.

The failure to recognize that the April/May 1990 event was reportable during the preparation of LER 92-005-00 was due to a lack of a questioning attitude on the part of the individuals researching the report.

CORRECTIVE ACTIONS

- In each event, repair of the leaking component was completed, components were tested, and leakage was then less than 1.34 gph.

- Regarding inadequate surveillance procedures, the monthly surveillance tests that measure external post accident system leakage (3PT-M16, 3PT-M18A, and 3PT-M18B) were revised. Revisions to these procedures were made previously (during 1993) such that the procedures now ensure that the cumulative external post accident system leakage (using the previous cumulative leakage in 3PT-C1 and the new leakage obtained in the monthly surveillance) is calculated for the operability determination of the monthly procedures. Therefore, leakage in excess of the technical specification leakage limit would be identified during the operability review of the surveillance tests rather than during the post-operability review of the tests. This ensures that external post accident leakage that is found to be in excess of the allowed limit while the plant is at power operation would be promptly identified.

- Currently, administrative controls are in place (3PT-C1 was revised on July 19, 1995) such that a new limit on external post accident system leakage which will ensure control room habitability is being used as the new operating limit.

- Regarding the apparent failure to recognize that the Technical Specification limit was not consistent with the design basis external post accident system leakage for control room habitability (1.34 gph), Technical Specification 4.4.I will be revised to reduce the maximum allowable leakage from external post accident systems to a value that will ensure post accident control room habitability. A proposed Technical Specification change will be submitted by April 30, 1996.

- Regarding the failure to identify that external post accident system leakage was in excess of the technical specification limit (2 gph) during the April/May 1990 event, a Technical Specification Interpretation (TSI) became effective in September 1992 which states that the 2 gph limit in Technical Specification 4.4.I is the maximum allowable leakage from RHR components <u>and</u> Safety Injection System components located outside containment and used during the recirculation phase of a design basis accident. This TSI clarifies the current technical specification wording. Further clarification of the technical specification wording may be required. As previously stated, a proposed Technical Specification change will be submitted by April 30, 1996.

• Regarding the failure to recognize that the control room habitability calculation established a new operating limit for external post accident system leakage, Procedure DCM-2 will be changed. The revision will require that when a calculation establishes a new operating limit, the new operating limit is identified and affected procedures and documentation that would require revision because of this new operating limit are identified. This revision will be completed by March 29, 1996.

• Regarding the lack of a questioning attitude on the part of the individuals researching LER 92-005-00, the Authority has extensively revised Administrative Procedure AP-8.2, "Deviation Event Analysis Manual," such that event investigations are now more thorough than they were in 1992.

ANALYSIS OF EVENT

These events are reportable under 10CFR50.73(a)(2)(i)(B) and 10CFR50.73(a)(2)(ii)(B); the plant was operated in a condition prohibited by the facility's Technical Specifications and in a condition that was outside the design basis of the plant. Technical Specification 4.4.I.2 limits maximum allowable leakage from the Residual Heat Removal System components located outside of the containment to 2 gallons per hour. Technical Specification 4.4.I.3 requires repairs or isolations to be made as required to maintain leakage within the acceptable criteria. As part of design basis reconstitution, the Authority has determined that in order to ensure post accident control room habitability, external post accident system leakage should be limited to 1.34 gallons per hour.

The Indian Point 3 systems that are monitored for leakage include post accident recirculation cooling and sampling systems external to the containment building. Leakage is identified, tracked, and totaled through the Indian Point 3 surveillance program.

A review of the safety injection pump maintenance history at the time of the original occurrence of the event in April 1992 did not identify recurring seal failures.

Two other events which had causes that were similar to the events described in this LER were identified. One event, described in LER 93-033-00, was similar in that technical specifications were inadequate. The other event, described in LER 93-016-00, was similar in that the design basis was not clearly defined.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 3	05000286	95	-- 016 --	00	7 OF 8

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

SAFETY SIGNIFICANCE

No safety consequences for the public health resulted from the events. No release of radiation occurred.

The Indian Point 3 Technical Specification basis states: "A recirculation system leakage of two gallons per hour will limit offsite exposure due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the design basis accident."

The leakages that were present during the three events described in this LER were evaluated. The evaluation assumed design basis fuel damage, maximally contaminated post-Loss-of-Coolant-Accident liquids, adverse meteorological conditions, and no PAB Filtration.

Using the 2.62 gph leakage in the April/May 1990 event as the bounding leakage, the potential doses to offsite receptors due to post accident containment leakage and leakage from Engineered Safety Feature components without filtration of airborne releases to the atmosphere from the Primary Auxiliary Building were calculated. It was concluded that, for such a scenario, all radiation exposures to offsite receptors would have been within the regulatory limits.

Although the worst case doses to control room personnel due to the three events described in this report would have exceeded the regulatory limit (General Design Criterion 19) if no action was taken to protect these personnel, the emergency plan ensures that action would be taken to protect these personnel.

The emergency plan establishes a control room health physics technician shortly after the initiation of the emergency plan. The control room health physics technician would be responsible for making recommendations to protect control room personnel from the effects of higher than expected radiation doses. If an accident did occur during the periods in which external post accident system leakage was above the design basis for control room habitability, the control room health physics technician would have been aware of the higher than expected control room radiation doses through control room surveys. Through these surveys, the control room health physics technician would be aware of the need for additional radiation protection for control room personnel. The emergency plan recommends use of potassium iodide when required for plant personnel. Additionally, self contained breathing apparatus is available for control room personnel.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit 3	05000286	95	-- 016 --	00	8 OF 8

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

The Authority has completed the Design Basis Documents for most of the safety related systems. The next part of design basis reconstitution will be a pilot program in which the Authority is currently planning to validate at least one system versus its Design Basis Document. This validation will determine the consistency between plant operating, surveillance, and maintenance activities and procedures and the as-built design basis information in the Design Basis Document. The pilot program is expected to be completed by December 1996.