Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
914 736.8001

New York Power Authority

L. M. Hill
Site Executive Officer

November 15, 1995
IPN-95-117

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 # 95-023-00
Isolation Valve Seal Water System in a Condition Prohibited by Technical Specifications due to Inadequate Modification Process

Dear Sir:

The attached Licensee Event Report (LER) 95-023-00 is hereby submitted as required by 10 CFR 50.73. This event is the type defined in 10CFR 50.73 (a) (2) (i) (B).

The Authority is making no new commitments in this letter.

Very truly yours,

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

LMH/vjw

Attachment

cc: See next page

IE22

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

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

INPO Record Center
700 Galleria Parkway
Atlanta, Georgia 30339-5957

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.

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TITLE (4) Isolation Valve Seal Water System in a condition prohibited by Technical Specifications due to Inadequate Modification Process.

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
10	16	95	95	-- 023 --	00	11	15	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	000

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)	X	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)
Jim Zach	(914)736-8038

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 October 16, 1995, with the reactor in the hot shutdown condition, Isolation Valve Seal Water (IVSW) system nitrogen header manual isolation valve (IV) 1492 was closed (as per drawing 9321-F-27463) and maintained closed per check-off list COL-CB-4. No procedural steps existed in System Operating Procedure SOP-CB-11 to open this valve post-accident to supply the required IVSW system nitrogen to the nine containment isolation valves (CIV) fed through this valve. On October 17, 1995, IVSW header manual isolation valve 1493 (water seal) was closed (as per drawing 9321-F-27463) and maintained closed per COL-CB-4. No procedural steps existed to open this valve post-accident to supply the required IVSW to the 13 CIVs fed through this valve. The cause of this error was an inadequate Modification Process. The Modification Process has since been improved. For corrective action, a Term Procedure Change (TPC) was written to SOP-CB-11 on October 17, 1995. This TPC added the necessary steps to open these valves.

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

DESCRIPTION OF EVENT

On October 16, 1995, at approximately 1720 hours with the reactor in the hot shutdown condition, (reactor power level 0, reactor coolant temperature at 230 degrees F, reactor coolant system pressure at 380 psig and pressurizer level at 29 percent), a system engineer, during his review of drawing 9321-F-27463 to evaluate the nitrogen supply to the IVSW system, noted the nitrogen valve IV 1492 was shown as closed with no guidance to open it in SOP-CB-11 for post-accident condition when this valve supplies nitrogen to nine CIVs. A Deviation/Event Report (DER) 95-2408 was written. On October 17, 1995, as part of the extent of condition review, IV 1493 was found in a similar condition. IV 1493 supplies IVSW to 13 CIVs. DER 95-2412 was written and a 48 hour LCO action statement was entered. Valves 1492 and 1493 were added to the IVSW system under a Mod 83-3-002 in 1983.

The IVSW system assures the effectiveness of the CIVs by providing a water seal (and in a few cases, a nitrogen gas seal) at the valves at a pressure greater than the containment accident analysis pressure. The system operates to limit the fission product release from the containment.

The accident analysis does not take credit for the operation of the IVSW system in the calculation of off-site accident doses. However, this system does provide assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

Most of the system operates automatically being actuated by a Phase A Containment Isolation Signal. Portions of the system are manually actuated. This includes thirteen IVSW valves (water seals) on Line no. 542 (IV 1493) and nine IVSW valves (gas seals) on Line no. 539 (IV 1492).

The system is controlled using SOP-CB-11, "Non-Automatic Containment Isolation." This procedure is called out in seven "EOPs" including ECA-2.1, "Uncontrolled Depressurization of All Steam Generators," ECA-3.1, "SGTR with Loss of Reactor Coolant - Sub-cooled Recovery Desired," ECA-3.2 "SGTR with Loss of Reactor Coolant - Saturated Recovery Desired," and ECA 3.3, "SGTR without Pressurizer Pressure Control." These four ECAs are procedures for events outside the design basis of the plant.

SOP-CB-11 is also called out in ES 1.2, ES 1.3, and E-3. The titles of those procedure are, "Post-LOCA Cooldown and Depressurization," "Transfer to Cold Leg Recirculation," and "Steam Generator Tube

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Rupture," respectively.

Modification 83-03-002 IVSWS was performed which among other things, added two valves: 1443B which isolated Line no. 539 (the valve was subsequently renumbered 1492) and 1400B which isolated Line no. 542 (the valve was subsequently renumbered 1493). Drawing 9321 F 27463-9, Revision 9, dated December 11, 1981, does not show these valves. Revision 11, dated December 3, 1984, added the two valves (1400B and 1443B), showing them open. Revision 13, dated January 14, 1986, renumbered valve 1443B to 1492 and 1400B to 1493 and the valves were still shown open. Revision 18, dated April 11, 1991, had the valves closed in accordance with a Technical Services Document and plant walkdown.

A review of the applicable Check Off List, COL-CB-4, "Isolation Valve Seal Water System," Revision 4 did not have the valves listed. Revision 5, (May 18, 1983) and Revision 6 (September 25, 1985) have valves 1400B and 1443B shut contrary to Revision 11 of the drawing which shows them open. Revision 10 of the COL (June 2, 1989) has the new valve numbers. It would appear the valves were left shut in accordance with the COL and contrary to the drawings (pre-1991). SOP-CB-11, Revision 2, dated January 16, 1982, did not include the valves. The modification package did not discuss the valves in the Nuclear Safety Evaluation or the Mod. They were apparently added through an Engineering Change Memo in 1983. In the distribution of the modification package, there was nothing highlighting the installation of the valves nor acknowledging implementation through documentation changes other than the drawings and check off lists.

CAUSE OF EVENT

The cause of the event was an inadequate Modification Process. It occurred in 1983 when the modification installed IV 1400B and 1443B (renumbered IV 1493 and 1492 respectively) through an Engineering Change Memo. The modification was not implemented properly by changing SOP-CB-11 to open the valves post-accident or to have them open normally on the check off list consistent with the drawing.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

CORRECTIVE ACTION

Term Procedure Change 95-1257 was written to SOP-CB-11, Revision 3, on October 17, 1995. This TPC added the necessary steps to open these valves.

The modification process, including the implementation through documentation update has been upgraded over the past ten years. The addition of a valve (such as 1492 or 1493) would now be implemented through proper procedure changes.

There have been many system walkdowns and procedure validations over the past few years to assure similar situations do not exist. These valves are a special case in that they are required to be manually repositioned and walkdowns have been done to assure normal component position and not to assume procedural guidance exists to change the position.

ANALYSIS OF EVENT

The lack of configuration control of valves IV 1492 and IV 1493 meant that up to nine (nitrogen) and thirteen (water seal) IVSW system valves respectively would not have had nitrogen or seal water supplied when required by emergency operating procedures. This is reportable under 10 CFR 50.73 (a) (2) (i)(B), "Any operation or condition prohibited by the plant's Technical Specifications." The absence of any procedural guidance to open these valves during post-accident condition would have prevented part of the IVSW system from performing its intended function and therefore be inoperable from 1983 through 1995. This condition is prohibited by the plant Technical Specifications above the cold shutdown condition beyond 7 consecutive days.

SAFETY SIGNIFICANCE

This event had no significant effect on the health and safety of the public. No credit is taken for the operation of the IVSW system to meet the requirements of 10 CFR 100 limits for the calculation of off-site doses in the plant design basis accident analysis. This is documented in FSAR Sections 6.5.1 and 14.3.5. The 22 valves on Line no. 542 and Line no. 539 were tested in accordance with Technical Specification 4.4.E.1.c and 4.4.E.2.c. Therefore, even without seal water or nitrogen post-accident because of IV 1492 and IV 1493 being closed, the isolation valve leakage would have been within the Technical Specification limits, or repaired if a problem was found.