Indian Point 3
Nuclear Power Plant
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New York Power Authority

John H. Garrity
Resident Manager

December 3, 1993
IPN-93-153

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop PI-137
Washington, D.C. 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Licensee Event Report # 93-047-00
"Improperly Configured Containment Isolation Valves, Caused By Personnel Error, Place The Plant Outside Design Basis"

Dear Sir:

The attached Licensee Event Report (LER) 93-047-00 is hereby submitted in accordance with the requirements of 10CFR50.73. This event is of the type defined in the requirements pursuant to 10CFR50.73(a)(2)(ii)(B). Also attached are the commitments made by the Authority in this LER.

Very Truly Yours,

JHGarrity

John H. Garrity
Resident Manager
Indian Point 3 Nuclear Power Plant

0800.4

JHG/vjm

cc: See Next Page

IE22

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U.S. NRC Resident Inspector's Office
Indian Point 3

Attachment 1
List of Commitments

Number	Commitment	Due
IPN-93-153-01	Corrective action for the incorrect orientation of valves CH-MOV-250A, B, C and D is currently under evaluation. LER 93-047 will be supplemented to identify the planned corrective action.	December 22, 1993
IPN-93-153-02	Containment isolation globe valves will be examined to determine whether they meet FSAR requirements for orientation.	January 14, 1994
IPN-93-153-03	Engineering will be counseled on the implications of developing a modification without assuring that all design criteria are met.	January 30, 1994
IPN-93-153-04	An independent design verification will be performed on the design criteria of the modification installing valves CH-MOV-250A, B, C and D. The balance of the valves installed by this modification will be examined to determine if they are in compliance with the modification design criteria.	January 14, 1994

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 6

TITLE (4)
Improperly Configured Containment Isolation Valves, Caused By Personnel Error, Place The Plant Outside Design Basis

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
11	03	93	93	-- 047 --	00	12	03	93	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)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	000		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)
James Sherman, Project Engineer	(914) 681-3293

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		12	22	93

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

On November 3, 1993, with the plant in cold shutdown and the reactor vented to atmosphere, Project Engineering Services concluded that the containment isolation valve configuration for the Chemical and Volume Control System (CVCS) Reactor Coolant Pump (RCP) seal water injection line was outside the licensing design basis. The condition was assumed to have existed since initial plant operation. The cause was personnel error of an indeterminate nature and inattention to detail. The installed configuration of Containment isolation globe valves will be examined to determine the extent of condition. The corrective action is being evaluated and will be identified in a supplement to this report.

DESCRIPTION OF EVENT

On November 3, 1993, at approximately 1000 hours, with the plant in cold shutdown (reactor power level at 12 cps, reactor coolant temperature at 102 degrees F, reactor coolant pressure at atmospheric and pressurizer level at 23%), Project Engineering Services (PES) concluded that the containment (NH) isolation valve (ISV) configuration for the Chemical and Volume Control System (CVCS) (CB) Reactor Coolant Pump (RCP) (P) seal water injection lines were outside the design basis in the Final Safety Analysis Report (FSAR). Deviation Event Report (DER) 677 was issued at approximately 1700 hours by PES.

As part of the program to address Generic Letter 89-10, a field walkdown was performed by the Operations and Maintenance Department. At that time, a discrepancy between the installed configuration of valves CH-MOV-250A, B, C & D and the requirements of FSAR Section 5.2.2 was identified. The discrepancy was tracked in the Authority's design basis document program as design basis document open item (DDOI) IP-3-CVCS-311-112, issued July 7, 1992.

The concern was researched by PES with the support of the Nuclear Engineering Department (NED) and the Nuclear Licensing Department (NLD).

Each of the four reactor coolant pump (RCP) seal water injection lines penetrating containment currently has two motor operated angle globe valves in series on the line outside the containment which are used as containment isolation valves. These lines supply water to the reactor coolant pump seals and, with the current valve orientation, normal flow direction is under the valve seats. Following a Design Basis Accident (DBA), the Isolation Valve Seal Water System (IVSWS) (BD) supplies water at a pressure slightly higher than the containment peak accident pressure to the leg of piping between the valves to act as a water seal.

The FSAR defines the isolation valves as class 3 (i.e., incoming lines with two manual isolation valves in series and manual seal water injection). Since the isolation valves are globe valves in series, the FSAR requires the valves to be oriented so that the IVSWS wets the valve stem packing. For IVSWS to wet the valve stem packing, the globe valves must be orientated so that the normal flow of water into the containment is under the valve seat of the outer isolation valve and over the valve seat of the inner isolation valve. This orientation assures that the seat of the inboard isolation valve is the first leakage barrier following the DBA. Because of their orientation, CVCS valves CH-MOV-250A, B, C & D, used as the inboard isolation valves, have their stem packing exposed to the environment in the leg of piping penetrating containment. This leg is not pressurized with water from the IVSWS and can be open to containment following a DBA.

PES reviewed the design history as documented on design drawings by both Westinghouse and United Engineers and Constructors Inc. and held discussions with Westinghouse. The review indicated that the valves were originally hand operated needle globe valves. The inboard needle globe isolation valves were changed to angle globe valves prior to initial operation. Documentation addressing the reason for the changes could not be retrieved to confirm the orientation. The CVCS drawing indicates that globe valves have flow under the seat except for 11 situations. The inboard isolation valves were not identified as exceptions so Project Engineering assumed that the valves were installed, prior to initial operation, in the current orientation.

After initial operation, a modification was made to address NUREG 0737 requirements. Two motor operated angle globe valves were added. The inboard containment isolation angle globe valves were replaced with motor operated angle globe valves. The modification did not indicate any change in orientation. The outboard motor operated angle globe valves were added between the original containment isolation valves and designated as the outboard containment isolation valves. The outboard motor operated angle globe valves were oriented with flow under the valve seat. PES concluded that the most probable cause of the event was original construction orienting the inboard angle globe isolation valve with flow under the valve seats. This reflected normal engineering practice. The implications for stem packing leakage were not recognized during the design review process or the subsequent modification to add motor operated valves.

CAUSE OF THE EVENT

The cause of the event was personnel error. Prior to initial operation, the error was associated with the engineering design and design review process of the Nuclear Steam Supply System supplier and Architect Engineer. The factors leading to the personnel error could not be identified because of the lack of documentation addressing changes to originally supplied equipment. For the subsequent modification, the error was made in the design change process. The factors leading to the error could not be identified by documentation. The PES review concluded that inattention to detail, a failure to check the FSAR requirements, was the most probable cause.

CORRECTIVE ACTIONS

To correct this event the following corrective actions will be performed:

- Corrective action for the incorrect orientation of valves CH-MOV-250A, B, C and D is currently under evaluation. LER 93-047 will be supplemented to identify the planned corrective action. The supplement is scheduled for December 22, 1993.

- Containment isolation globe valves will be examined to determine whether they meet FSAR requirements for orientation. This evaluation is scheduled for completion by January 14, 1994.

- Engineering will be counseled on the implications of developing a modification without assuring that all design criteria are met. Counseling is scheduled for completion by January 30, 1994.

- An independent design verification will be performed on the design criteria of the modification installing valves CH-MOV-250A, B, C and D. The balance of the valves installed by this modification will be examined to determine if they are in compliance with the modification design criteria. This is scheduled for January 14, 1994.

No additional corrective action is required to prevent recurrence of this event. The event probably occurred during initial design and construction and was not discovered during a subsequent modification. It could also have occurred during the subsequent plant modification. In either case, the current plant modification process is quite different and requires consideration of FSAR commitments. The current modification process will prevent recurrence.

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

ANALYSIS OF THE EVENT

This event is reportable under 10 CFR 50.73(a)(2)(ii)(b). The Licensee shall report any event or condition that resulted in the plant being in a condition that was outside the licensing design basis of the plant. For containment isolation lines isolated by globe valves and provided with seal water, the FSAR requires valves to be oriented so that the seal water wets the stem packing. The resulting water seal will block leakage through the valve stem during a Design Basis Accident. The RCP seal water line inboard containment isolation valves, CVCS valves CH-MOV-250 A, B, C & D, have not met this licensing design basis since initial plant startup because they were installed in the wrong orientation.

Recent events have been identified as similar because they report: design or construction errors that occurred prior to startup, LERs 93-045, 93-044, 93-043, 93-036, 93-035, 93-030 and 93-026 ; events identified during the resolution of DDOIs, LERs 93-045, 93-044 and 92-006; and, events that affect containment leakage requirements, LERs 93-043, 93-035, 93-016, 93-012 and 93-002.

SAFETY SIGNIFICANCE

There is no effect on the public health and safety from this event. This conclusion is based upon the following considerations:

- The valve stem and packing of valves CH-MOV-250 A, B, C and D are designed to be backseated. IP3 does not backseat motor operated valves without an engineering evaluation. These valves are not backseated during normal operation so the valves are exposed to CVCS system pressure, greater than primary system pressure. The valves are located in the plant auxiliary building valve gallery which is not heavily traveled because it is a radioactive control area. Significant degradation of packing and the associated leakage would be detected and corrected during periodic visits. Because of the significant difference between the operating pressure and the post accident pressure, minimal leakage is expected.

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

- Leak rate testing of similar valves identifies no significant packing leakage. The isolation valves for the four penetrations are leak rate tested using procedure 3PT-R25. Testing is performed by pressurizing between the valve seats. Testing does not test leakage through the stem packing of valves CH-MOV-250 A, B, C and D but does pressurize the stem packing of the outboard isolation valves which are of the same design. No significant leakage was detected so there could be no significant leakage through the stem packing.

- Containment leak rate testing does not act to test the leakage through the stem packing of valves CH-MOV-250 A, B, C and D but does provide reasonable assurance that the containment design leak rate will be met. During testing the RCP seal water line is drained and vented but the isolation valve stem packing does not see the test pressure since there is a check valve (i.e., 251E, F, G and H for valves CH-MOV-250 A, B, C & D, respectively) between the valve stem packing and the vent to containment atmosphere. The Loss of Coolant Accident (LOCA) is assumed to cause the same configuration by breaking the seal water line inside the shield wall with the two additional check valves located there. The break opens the line to containment atmosphere. There is reasonable assurance that the containment design leakage rate will be met since the configuration during LOCA and testing is the same. The potential for substantial leakage through the check valve and packing would be detected during containment testing.

- Small break LOCA does not present a stem packing bypass leakage concern. RCP seal water is injected into the reactor coolant pumps between the thermal barrier and the shaft seal. The injected seal water flow becomes shaft seal leakage or enters the Reactor Coolant System through the RCP pump shaft labyrinth seal. RCP seal degradation following a small break LOCA (or other event) would not cause reverse flow since there are three check valves between each RCP seal and the valve stem packing of the associated isolation valve, CH-MOV-250 A, B, C or D.

To evaluate the extent of condition, containment isolation globe valves which are supplied with IVSWS will be examined to determine whether they meet FSAR requirements for orientation.