Indian Point 3
Nuclear Power Plant
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NewYorkPower Authority

Robert J. Barrett
Site Executive Officer

December 18 , 1998
IPN-98-140

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 # 1998-009-00,
"Plant Shutdown Due to Inoperable Containment Isolation Valves Caused by Improper Surveillance Procedures."

Dear Sir:

The attached Licensee Event Report (LER) 1998-009-00 is hereby submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73 (a)(2)(i)(A) for completion of a plant shutdown required by Technical Specification. This event is also of the type defined in 10CFR50.73 (a)(2)(i)(B) for conditions contrary to the Technical Specifications.

The Authority is making no new commitments in this LER.

Very truly yours,

Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant

IE22

cc: See next page

cc: Mr. Hubert J. Miller
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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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point 3	05000286	1 OF 6

TITLE (4)

Plant Shutdown Due To Inoperable Containment Isolation Valves Caused by Improper Surveillance Procedures

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
11	18	1998	1998	-- 009 --	00	12	18	1998	N/A	05000
									FACILITY NAME	DOCKET NUMBER
									N/A	05000

OPERATING MODE (9)	Pwr
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)
Patric Conroy, Supervisory System Engineer	(914) 736-8305

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
D	AB	ISV	W315	Y					
A	CC	ISV	C684	Y					

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 November 18, 1998, at 2350 hours, a plant shutdown was commenced in accordance with Technical Specification(TS) 3.6.A, "Containment Integrity," for inoperable containment isolation valves (CIV) due to inadequate testing in the past. The procedures for valve leak rate testing did not specify making adjustments to the test pressure to account for the elevation of the test rig relative to the elevation of the valve being tested or for the head of water or system pressure on the other side of the valve. Also, the procedures did not specify draining water within the test boundary when testing with gas as a medium. Ninety-five valves were identified for retesting. Within this grouping, eighty-two valves would require leak testing to ensure the proper delta pressure across the valve seat. The other thirteen valves were previously tested with nitrogen over water (a mixed medium). Two CIVs were found to be inoperable due to past maintenance and improper testing, one valve was previously identified that led to the discovery of this event and one valve was found after the plant shutdown and during the re-testing. The cause of the event was that the original and subsequent revisions of the test procedures were inadequate. Corrective actions included repairing the two valves, making the improper procedures inactive and writing specific procedures for retesting eighty-eight valves, obtaining a one-time TS change for seven valves that do not need to be tested until the next refueling outage, and performing an extent of condition review of other leak rate testing and other surveillance procedures. Before the next planned outage, the maintenance and test procedures will be revised with engineering review of the Appendix J test program for technical adequacy. This event did not have a risk significant effect on the health and safety of the public based on the safety assessment from both a deterministic and probabilistic viewpoint.

Note: The Energy Industry Identification System Codes are identified with brackets { }.

Description of the Event

On November 18, 1998, at 2350 hours, with reactor power at 100 percent, a plant shutdown was commenced in accordance with Technical Specification(TS) 3.6.A, "Containment Integrity," for inoperable containment {NH} isolation valves {ISV}. They were declared inoperable when it was determined that past TS required testing of containment isolation valves was inadequate. A test procedure (3PT-R25, "Isolation Valve Seal Water System Test"), performed in August of 1997 did not require the piping upstream and downstream of valves being tested to be vented and depressurized or the test pressure increased to compensate for system pressures and placement of the test rig. Specifically, past testing of Component Cooling Water (CCW) {CC} containment isolation valves on lines for cooling auxiliaries to the Reactor Coolant Pumps (RCP) {AB} did not account for the CCW system pressure that was present at the time of testing. In accordance with 10CFR50.72(b)(1)(i)(A) a one-hour non-emergency report (ENS#35056) was submitted to the NRC for the TS required shutdown.

The inadequate testing was discovered during an equipment failure evaluation (EFE) for a containment isolation valve (SP-AOV-956C) that was discovered to be inoperable on November 1, 1998. This valve was discovered to leak by in the closed position. The EFE found that the valve's stem travel was improperly adjusted so the valve would not fully seat. The improper adjustment occurred while performing actuator refurbishment and spring replacement that was completed on August 14, 1997. As part of the EFE, an adjustment of the stem travel with the packing installed was unsuccessful. It was determined that the packing hinders the ability to determine if the valve seated and may have played a role in preventing a successful stroke adjustment. The maintenance procedure (VLV-032-GEN) was inadequate in that it did not have instructions to ensure valve-packing loads are not present when making valve stem adjustments. Later, as part of the EFE and specific to this valve failure, an extent of condition review was done for similar valves that may not have been properly adjusted and it was determined no further action is required beyond the re-testing discussed later in this LER. Also, a review of similar valve maintenance procedures found no problems or issues with regard to making valve stem adjustments. In August 1997, the leak rate test (3PT-R25) was credited as part of the post-maintenance test and should have discovered the valve's inability to isolate. The EFE determined that the upstream side of the valve was not properly vented, thereby precluding accurate test results that would have identified the inoperable valve. After November 1, 1998, SP-AOV-956C was replaced with a new valve that was properly leak tested using a specific test (ENG-625).

As part of the investigation for this shutdown event, an extent of condition review was performed to review Appendix J, Type B & C testing and IST valve leakage testing in order to determine which valves required leak rate re-testing. It was identified that past leak rate tests (3PT-R25 and 3PT-R35 series) were inadequate. The procedures did not specify making adjustments to the test pressure to account for the elevation of the test rig relative to the elevation of the valve being tested or for the head of water or system pressure on the other side of the valve. Also, they did not specify draining water within the test boundary when testing with gas as a medium.

Based on the review of documentation of past tests and plant documents, ninety-five valves were identified for retesting. Within this grouping, eighty-two valves would require leak testing to ensure the proper delta pressure across the valve seat in order to meet TS 4.4.E.2 and TS 4.4.E.3. TS sections 4.4.E.2 and 4.4.E.3 require testing of containment isolation valves that are sealed with water from the Isolation Valve Seal Water System (IVSWS) {BD} and Service Water System {BI}, respectively, to quantify the leakage rate of water when pressurized $\geq$ 1.1 Pa, where Pa is the calculated peak accident pressure. The remaining thirteen valves, although their last test was satisfactory for delta pressure, were tested with nitrogen over water (a mixed medium) contrary to TS Table 4.4-1, that requires nitrogen as a test medium. TS section 4.4.E.1 requires verification that the combined leakage rate for containment bypass leakage paths that are not sealed by qualified seal systems to be $\leq$ 0.6La when pressurized at $\geq$ Pa, where La is the maximum allowed accident leak rate. The nitrogen portion of the IVSWS system is used post-accident to pressurize the liquid in the lines between containment isolation valves and between the discs of double disc valves to about 225 psig, which is in excess of recirculation fluid pressure on one side of the valves. Of these thirteen valves subject to the mixed testing medium, plans included re-testing six valves. The remaining seven valves could not be tested without draining the high head safety injection {BQ} normal suction path, without increasing the risk of interrupting core cooling, without stopping core cooling or possibly requiring a full core offload. These seven valves were evaluated by Engineering and it was determined that the associated lines would be filled with water for thirty-days following a Design Basis Accident (DBA) and do not provide a containment atmosphere pathway when considering the most limiting single active failure. Based on the engineering evaluation, a one time Technical Specification Amendment was submitted and approved (NRC Letter dated November 27, 1998).

Engineering prescribed test requirements to ensure the proper delta pressure across the valves' seat while considering the system pressures, placement of the test rig, height of water, proper venting and draining. Specific procedures (ENG-626, 627, 629, 630, 631, 632, 633, 635, 636, 3PT-R35A&B) were developed or revised and testing commenced. Satisfactory results were achieved, except for one valve (AC-MOV-797). On November 27, 1998, the re-testing identified a CCW containment isolation valve (AC-MOV-797) in the line for cooling RCP auxiliaries failed to meet operability criteria due to excessive leakage (DER-98-2315). After further investigation, it was determined to be a pre-existing condition and on November 28, at 1434 hours, a four-hour non-emergency report (ENS#35085) to the NRC was completed in accordance with 10CFR50.72(b)(2)(i) for a degraded condition found while shutdown. Subsequently, it was determined that this condition (inoperability of one of two series containment isolation valves) does not meet the threshold for making a four-hour report, but does require reporting the event in a LER, as reported in this LER.

Upon disassembly of Valve AC-MOV-797, it was determined that the valve wedge did not seat properly into the seat. The valve required machining of six thousandths of an inch from each side of the wedge in order to obtain the proper seating surface as confirmed from blue checking. A review of this valve's leakage history showed that the valve historically had leakage and in 1989 it was deemed excessive. Work history included a few times that this valve was disassembled, cleaned, inspected, lapped and blue checked and reassembled with satisfactory results documented. Post maintenance testing included the TS required leakage testing using the test method of 3PT-R25, which as a result of this event was determined to be inadequate.

A review of 3PT-R25 procedure revisions since 1978 revealed that the original and subsequent revisions did not have adequate instructions to ensure the proper delta pressure across the valve seats while testing. The original procedure through revision two provided a statement that allowed pressure in the systems outside the test boundary up to calculated accident pressure while testing the valves. But in 1981 for one completed test, the test performer recognized and documented the need to adjust the test pressure to compensate for the system pressure in order to achieve the proper delta pressure. The subsequent revisions from 1987 were silent on this subject of line pressure outside the test boundary. Procedure steps for testing one set of valves that are supplied with IVSWS nitrogen required an adjustment to the test pressure based on the pressure outside the test boundary, but overall adequate venting and draining were not prescribed for valves supplied by either IVSWS water or nitrogen. Interviews with plant operators who performed some of these past tests identified that some degree of knowledge existed to ensure the required delta pressure was achieved for earlier testing but later test performers may not have had that knowledge and that knowledge was never captured into the procedure.

As part of the investigation, an extent of condition review was performed on other Technical Specification (TS) surveillance procedures. Personnel from Engineering, Operations, Instrumentation and Control, and Performance reviewed a sample of selected TS procedures. The selection of tests to review was based upon tests that may have results influenced by placement of test equipment, temperature and pressure within or outside test boundaries, and line losses. Questions were raised during the review and upon evaluation by engineers no re-testing was required.

Cause

The cause of the plant shutdown was inadequate Technical Specification required leak rate testing of containment isolation valves. The cause for not adequately testing containment isolation valves is that the original and subsequent revisions of the Appendix J valve leakage test procedures (3PT-R25 and 3PT-R35 series) were inadequate.

Corrective Actions

Corrective actions included re-testing eighty-eight valves with test methods that were appropriate for the plant conditions.

A one time Technical Specification Amendment was submitted and approved to not require re-testing of seven specific valves until the next refueling outage (NRC Letter dated November 27, 1998).

Valve SP-AOV-956C was replaced and valve AC-MOV-797 was repaired.

Maintenance procedure VLV-032-GEN will be revised prior to its next planned use and by January 24, 1999, to ensure that valve packing loads are not present when making valve stem adjustments.

An extent of condition review for leak rate testing was completed as part of this event and as described above. An extent of condition review of other surveillance test procedures was completed as part of this event and as described above.

Engineering will review for technical adequacy the Containment Leak Rate Testing Program (Appendix J testing). This is planned to be done by May 1, 1999 in order to ensure adequate time to revise the Appendix J procedures before the next refueling outage which is planned for September, 1999.

Operations made inactive the inadequate Appendix J valve leak rate test procedures (3PT-R25 & 3PT-R35 series) until they can revise them, with lessons learned from this event, before the next refueling outage scheduled for September, 1999.

Analysis of the Event

This event is reportable under 10CFR50.73(a)(2)(i)(A), for completion of a Technical Specification (TS) required shutdown. Hot shutdown was achieved on November 19, 1998.

The event is also reportable under 10CFR50.73(a)(2)(i)(B), for conditions contrary to TS. Ninety-five containment isolation valves were identified for retesting and thereby were considered not tested in accordance with Technical Specification 4.4.E. The duration of the inadequate testing was from at least August 1997 until plant shutdown to cold shutdown condition on November 20, 1998. The two valves that failed (AC-MOV-797 & SP-AOV-956C) were inoperable while the plant was above cold shutdown, which is a condition prohibited by TS 3.6.A, "Containment Integrity." The duration of inoperability of valve 956C was from August 1997 to November 1, 1998. The duration of inoperability of valve 797 could have been from 1989 when excessive leakage was identified until plant shutdown to cold shutdown condition on November 20, 1998. In the above cases, the testing documentation does not provide sufficient evidence to conclude that previous tests compensated for influences by the test configurations or system pressures and therefore it is indeterminate if these conditions existed previously to what is stated above.

A review of the LERs for the past five years did not identify similar events where Appendix J or IST valve leakage testing was inadequate.

Safety Significance

This event did not have a risk significant effect on the health and safety of the public based on the following assessment from both a deterministic and probabilistic viewpoint. From a deterministic viewpoint the following assessment credits the redundant in-series valve that would have performed the isolation function or, if these redundant valves failed as a result of single failure, the IVSWS or CCW systems would provide, in whole or part, a seal water function. From the probabilistic viewpoint, the following assessment concludes that this event had a negligible increase to the Large Early Release Frequency (LERF).

Deterministic Safety Assessment

For the inoperability of SP-AOV-956C and AC-MOV-797, their respective in-series redundant valves SP-AOV-956D and AC-MOV-769 would have performed the containment isolation function.

For the single failure case of SP-AOV-956D, associated with the in-series inoperable valve SP-AOV-956C, the IVSWS provides automatic seal water, thereby mitigating any potential release from this path.

For the single failure case of AC-MOV-769, associated with the in-series inoperable valve AC-MOV-797, with offsite power available during the DBA, the CCW system would be in operation and in effect perform like a seal water system, maintaining pressure across the closed but not leak tight valve seat of AC-MOV-797. Because the CCW pump pressure is higher than containment accident pressure and the as found condition of AC-MOV-797 was closed but leaking, the in-leakage of CCW into containment could be expected to be addressed as a long term (i.e., at least several hours) remedial issue during the postulated event. This is based on a conservative estimate of the predicted leakage of CCW fluid under such conditions. Isolation of one of two CCW headers would be considered to address the in-leakage since the plant systems are designed such that safe shutdown can be maintained with one CCW header.

For the single failure case of AC-MOV-769, associated with the in-series inoperable valve AC-MOV-797, with a loss of offsite power, the CCW system would not be in operation during the injection phase of the DBA. The CCW piping configuration is such that check valve AC-770 would provide an additional barrier and the overall CCW system piping configuration is such that it would act to some degree as a water seal to restrict the postulated containment atmospheric path. AC-770 is a qualified valve except that its leak rate is not tested. Assuming leakage through AC-770 based on other similar swing check valve historical leakage results times a factor of ten, combined with the leakage through AC-MOV-797, the CCW system would provide some degree of back pressure due to its loop seal configuration and again a long term measure (i.e., at least several hours), such as isolating the header, could be taken if required.

For the single failure of either SP-AOV-956D or AC-MOV-769 to close, the IVSWS inventory would be within its twenty-four-hour design capability based on the current total measured leakage of the other IVSWS fed valves and the IVSWS flow rates assumed for the subject sampling and CCW process lines (i.e., the inoperable SP-AOV-956C and AC-MOV-797 valves coupled with the assumed single failure of either SP-AOV-956D or AC-MOV-769). For the limiting IVSWS prescribed single failure of the largest valve fed with IVSWS (i.e., an eight inch valve), the IVSWS inventory would be slightly below its twenty-four design capability when considering the inoperable valves SP-AOV-956C and AC-MOV-797. In this case, the IVSWS tank low level alarm that normally prompts the operators to align primary water for replenishment would be expected to occur slightly earlier.

Probabilistic Safety Assessment

The LERF was reviewed considering the inoperability of SP-AOV-956C and AC-MOV-797, and it was determined this condition has a negligible impact on the LERF. For the inoperability of SP-AOV-956C, which provides isolation of a 3/8-inch line, the size of the line precludes the possibility of having a LERF (i.e., the IPE defines a "large" release as greater than 2" in diameter). For the inoperability of AC-MOV-797, when considering the frequency of a large break LOCA that is assumed to result in a pipe whip/missile that ruptures the CCW line to the RCPs, and the probabilities of valve AC-770 and AC-MOV-769 failing to close, with the probability of core damage from a LBLOCA, the frequency of a LERF is 2.3E-10/yr. Since increases less than 1.0E-7/yr are considered to be negligible, this worst case event is considered non-risk significant.