Indian Point 3
Nuclear Power Plant
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New York Power Authority

Robert J. Barrett
Site Executive Officer

August 25 1999
IPN-99-093

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 # 1999-009-00
A Condition Prohibited by Technical Specifications Because the Allowed Outage Time was Exceeded for a Loss of Containment Integrity Caused by Personnel Error and Inadequate Procedures

Dear Sir:

The attached Licensee Event Report (LER) 1999-009-00 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)(B) and (a)(2)(ii)(B).

The Authority is making no new commitments in this LER.

Very truly yours,

Robert J. Barrett

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

cc: See next page

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PDR ADOCK 05000286
S PDR

cc: Mr. Hubert J. Miller
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Indian Point 3 Nuclear Power Plant

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
A Condition Prohibited by Technical Specifications Because the Allowed Outage Time was Exceeded for a Loss of Containment Integrity Caused by Personnel Error and Inadequate 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 05000
07	27	1999	1999	-- 009 --	00	08	25	1999	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)	100	20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
		20.2203(a)(1)	20.2203(a)(3)(i)	X 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)
Marie Gillman, Shift Manager	(914) 736-6244

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

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On July 27, 1999, operations discovered that a modification, which installed a new Isolation Valve Seal Water System (IVSWS) test connection between two containment isolation valves (CIVs) (WD-AOV-1728 and WD-AOV-1723), was improperly installed and improperly returned to service. The installed configuration allowed a pathway from containment. This condition could have resulted in exceeding 10 CFR 100 and 10 CFR 50, Appendix A, GDC-19 dose limits under a Design Basis Accident and a single failure of CIV WD-AOV-1728 to close on demand. This condition existed for greater than the Technical Specification allowed outage time. The cause of the event was personnel error due to inattention to detail and an inadequate work control process regarding system restoration after performing work. Corrective actions included counseling appropriate personnel on management's expectations for attention to detail and the need to perform adequate error detection, revision of the work control (WC) process on requirements for a recovery plan, and an extent of condition review of work packages (WP). WPs for modifications and maintenance will be reviewed against the revised WC process prior to system restoration. A task analysis will be performed on the operations impact process and lesson learned implemented as required. The event had no actual significant effect on public health and safety. This event did not qualify as a safety system functional failure.

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

Note: The Energy Industry identification system Codes are identified within the brackets {}

DESCRIPTION OF EVENT

On July 27 , 1999, at approximately 1725 hours, with steady state reactor power at approximately 100 percent, the Construction Services Manager was performing management observations and discovered water on the floor of the Primary Auxiliary Building (PAB) {NF}, at elevation 41 feet. The manager notified a Nuclear Plant Operator (NPO) via Health Physics (HP). The NPO's investigation determined the water was leaking from the Isolation Valve Seal Water System (IVSWS) {BD}. The leak was from a new IVSWS test connection valve (IV-1646) installed by a modification between containment isolation valves (CIV) {JM} WD-AOV-1728 and WD-AOV-1723 on the containment sump pump {P} discharge line. At approximately 1727 hours, Operations closed the normally open CIVs, turned the sump pump control power off, and entered a one hour Technical Specification Limiting Condition for Operation (LCO) for containment integrity. Waste Management personnel estimated the quantity of water that leaked to be approximately 40 to 50 gallons. A contamination smear survey by HP showed negligible contamination. Operations review of the containment sump pump integrator recording {FQR} showed no change during the event and that the pump remained off. Investigation discovered that all the newly installed test valves {TV} were open and IVSWS valve IV-1646 had a fitting instead of a cap installed on its discharge side. At approximately 1730 hours, Operations closed all the newly installed IVSW test connection valves. At approximately 1810 hours, Operations de-activated CIVs WD-AOV-1728 and 1723, verified closed manual valves SA-24-1 and SI-859, closed and de-activated valves DW-AOV-1 and 2, RC-AOV-519 and 552, and exited the one hour LCO for containment integrity. A deviation event report (DER 99-01517) recorded the condition and investigations initiated. Further investigation determined that CIV WD-AOV-1728 was re-activated shortly after 1448 hours and later opened without making the new IVSWS test connection operable.

At approximately 1900 hours, Operations determined the plant was potentially outside its design basis from shortly after 1448 hours, when CIV WD-AOV-1728 was activated and later opened, until 1810 hours, when CIV WD-AOV-1728 was closed and de-activated and containment integrity re-established for the containment sump piping penetrations {PEN}. During the time the containment sump pump piping penetrations were unisolated, a single failure of the normally open CIV WD-AOV-1728 to close during a Design Basis Accident (DBA) could result in an open pathway from containment atmosphere to the PAB via an open uncapped 3/8 inch IVSW test connection valve IV-1646. At 1957 hours, a one hour non-emergency notification report (Log no. 35965) was made to the NRC for the plant potentially in a condition outside design basis.

The containment sump pump discharge line is a two inch diameter pipe {PSP} to the Waste Holdup Tank of the liquid waste processing system {WD} that has two normally open air operated CIVs outside containment which close on a Phase A containment isolation signal {JE}.

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

The IVSWS is designed to provide water or gas seals to certain CIVs and into piping between CIVs to assure the effectiveness of containment isolation in lines which are either connected to the reactor coolant system (RCS) {AB}, or could be exposed to the containment {NH} atmosphere during any condition which requires containment isolation. IVSWS is designed to inject pressurized water or nitrogen to maintain a seal between the two isolation points. The water is injected at a slightly higher pressure than containment design pressure resulting in a water seal that blocks leakage of the containment atmosphere. IVSWS consists of a nitrogen supply bank and a closed water tank {TK} in which the valve sealing fluid is stored under nitrogen gas pressure, and a network of small tubing {TBG} lines for distributing the sealing fluid to the CIVs. System operation is initiated by any automatic safety injection signal or manually.

The modification of the IVSWS was to replace the existing IVSWS line and valve connected to the sump pump discharge line between its two isolation valves, with new prefabricated and tested lines, valves and fittings. The IVSWS lines are approximately 3/8 inch outside diameter (OD) tubing. A work package was developed to install the modification which included an operational assessment of the impact of implementing the modification. The Operational Impact Sheet was developed by an unlicensed Nuclear Plant Operator (NPO) and included preparation of LCOs, potential LCOs (PLCO), and Protective Tagging Orders (PTOs). The worksheet identified that LCOs were required for IVSWS and fire protection, and that a PTO was required for personal protection. The worksheet failed to identify that a PLCO should be entered for containment integrity. A licensed operator performed an assessment for operational impact but failed to identify all of the containment integrity issues and the need to enter a potential LCO and maintain the CIVs closed and de-activated until the new IVSWS test connections were returned to operable.

On July 26, at approximately 0400 hours, Technical Specification 3.3.C.2.A was entered with a seven (7) day LCO for the IVSWS Station 2 Header, to begin installation. Operations began to apply PTO 99-1047 for IVSWS work at 0426 hours, which was completed at 0510 hours. Installation was performed per associated work packages and after completion Operations authorized the removal of the PTO on July 27, at 1448 hours. PTO restoration was performed in accordance with established procedures which returned the system to the pre-PTO positions for the components. Clearing of the PTO began shortly after PTO removal authorization. At PTO step 3 the air supply to CIV WD-AOV-1728 was opened, then 15 steps later the control switch for CIV WD-AOV-1728 was opened. Subsequently, shortly before 1610 hours, the automatic IVSWS Station No. 2 Header isolation valve IV-1407 was locked open. The PTO removal was completed at 1645 hours. At 1645 hours, Operations was briefed on the IVSWS Modification Acceptance Test (MAT) by the test group supervisor. Modification turnover documentation was to follow a successful MAT. The PTO removal re-activated the closed CIVs to their normally open position prior to testing the IVSWS for operability. A PLCO was entered at 1810 hours for containment integrity, and a PTO and Caution Tagging Order (CTO) applied for testing the new installation (MAT) and to ensure the new IVSWS test connection valves are closed and capped.

After installing the proper IVSWS test connection cap on valve IV-1646 and performing a satisfactory MAT, the PLCO for containment integrity on the sump pump discharge line was exited and the CIVs declared operable at 1632 hours, on July 28, 1999. An extent of condition review was performed on other completed modification work packages associated with IVSWS modifications. These work packages (IVSWS modifications) were reviewed and determined that containment integrity was not violated. Additionally, all work packages will be reviewed using the revised work control process upon return to work control prior to system restoration.

CAUSE OF EVENT

The cause of the failure to ensure the IVSWS modification was operable prior to activating and opening the CIVs was personnel error. The personnel error was due to cognitive errors as a result of poor work practices and a defective procedure due to a lack of adequate written instructions/omission of information. The work control process did not contain a method for providing requirements for maintaining configuration control and restoration of systems to operability following installations, but prior to modification turnover to Operations. Recovery activities relies on the skill and knowledge of the preparer, reviewer, and Field Support Supervisor (FSS). Operations failed to adequately plan for recovery of the impacted systems during the time period after modification installation (clearance of work package) and completion of the modification turnover documentation. The modification was not installed in accordance with the work package, but the CIVs were activated and opened in accordance with the PTO removal steps prior to the Modification Acceptance Test (MAT) and Modification Turnover Documentation (MTD) completion. The failure to identify the impact of the work actions on containment integrity during completion of the operations impact sheet was due to poor work practices. Self checking and attention to detail were inadequate.

A licensed operator performed an assessment for operational impact but failed to identify all of the containment integrity issues and the need to enter a potential LCO and maintain the CIVs closed and de-activated until the new IVSWS test connections were returned to operable.

The cause of the failure to install a cap instead of a fitting on the outlet of IVSWS valve IV-1646 could not be determined but was likely due to poor work practices as a result of inattention to detail. The IVSWS modification was fabricated, bench tested and turned over, to Construction Services for installation. At some time a fitting instead of a cap was installed on valve IV-1646. The work package was signed-off as completed per the modification package.

Failure of installation personnel (non-licensed contractor and utility workers) to identify the missing valve cap, after signoff of the work package verifying the installation was in accordance with the modification package and drawings, was caused by inattention to detail. Contributing causes were environmental conditions that included uncomfortable temperature and humidity at the work site and the orientation of valve IV-1646 and its fitting making it difficult to view that a Swagelok fitting verses a Swagelok cap was installed.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Authority's corrective action program to address the causes of this event.

- The appropriate personnel were counseled on management's expectations for attention to detail and the need to perform adequate error detection.
- The work control procedure (SPO-SD-01, Work Control Process) was revised to incorporate lessons learned (included a second licensed operator's review of the LCO/PLCO issues for a work package and a check on the Operations Impact Sheet for the necessity of a recovery plan).
- Procedure CON-AD-01, "Control of General Maintenance Contractors and Other Contractor Work Efforts," was revised by adding a requirement for an Authority representative to walk-down a job upon completion.
- An extent of condition review was performed on other completed work packages associated with IVSWS modifications. There were no other instances identified in these work packages where containment integrity was violated.
- An inspection was performed of the remaining prefabricated units that revealed two additional assembly discrepancies. In one unit a cap was missing and in another unit a cap and fitting were switched. The identified discrepancies were corrected.
- Work packages associated with modifications and maintenance will be reviewed against the revised work control process to determine if restoration steps need to be included. The review will be performed when the work package is returned to work control prior to system restoration.
- A task analysis will be performed of the operations impact process and lesson learned implemented as required. The analysis is scheduled to be completed by February 16, 2000.

ANALYSIS OF EVENT

The event is reportable under 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(ii)(B). The licensee shall report any operation or condition prohibited by the plant's Technical Specifications (TS) and any event or condition that resulted in the plant being in a condition that was outside the design basis of the plant.

The event meets the reporting criteria for a condition prohibited by TS because TS 3.6.A.1 specifies that containment integrity shall not be violated unless the reactor is in the cold shutdown condition. TS 3.6.A.3 further specifies that if containment integrity requirements are not met when the reactor is above cold shutdown, containment integrity shall be restored within one hour.

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When the connective piping, replacement valves, and the modification to the IVSWS were not tested to demonstrate operability before CIVs WD-AOV-1728 and WD-AOV-1723 were re-activated and opened, containment integrity was violated. This condition existed from shortly after 1448 hours when CIV WD-AOV-1728 (inner CIV) was re-activated to approximately 1810 hours when CIV WD-AOV-1728 was de-activated and closed, a period of approximately 3 hours and 22 minutes. Because the condition exceeded the TS one hour LCO Allow Outage Time (AOT) for containment integrity, the plant was in a condition prohibited by TS for containment integrity . The Standard Technical Specifications (STS) contains an allowed outage time of four (4) hours for this condition.

This event meets the reporting criteria for a condition outside the design basis of the plant because from shortly after 1448 hours, when CIV WD-AOV-1728 was re-activated and opened, until 1810 hours, when CIV WD-AOV-1728 was closed and de-activated and containment integrity re-established for the containment sump piping penetrations, a single failure of the normally open CIV WD-AOV-1728 during a DBA could result in an open pathway from containment atmosphere to the PAB via an open and uncapped IVSWS valve IV-1646. The resultant potential release of DBA containment atmosphere could have exceeded 10 CFR 100 and 10 CFR 50, Appendix A, GDC-19 limits.

A review of the past two years of Licensee Event Reports (LERs) for events that involved loss of containment integrity due to inadequate testing identified LER 99-001, and LER 98-009. Review of LERs since January 1998 for events whose cause was inadequate work processes did not identify any LERs that were applicable. Corrective actions for LERs 99-001 and 98-009 failed to prevent this event because they were for testing procedure inadequacies (corrective actions were for inadequate testing procedures that failed to require proper venting or accounting of system pressures), whereas this event was for inadequate assessment of the impact of the modification and inadequate written instruction for the required actions to return the system to service. The testing procedures for containment integrity for this event were adequate therefore the corrective actions for the previous LERs that revised testing procedures would not have prevented this event.

SAFETY SIGNIFICANCE

This event had no actual significant effect on the health and safety of the public.

Review of this event against the guidelines of Nuclear Energy Institute (NEI) 99-02 Draft Rev. B, "Regulatory Assessment Performance Indicator Guideline," concluded it was a possible candidate as a safety system functional failure (SSFF) for containment integrity. Further assessment using the NEI guidelines determined that the event would not be considered a SSFF because there would be no failure of the capability to isolate containment. In accordance with the NEI guideline, it is not necessary to consider a single random failure of the CIV WD-AOV-1728 to close on a Phase A isolation signal, absent an identified potential failure mechanism.

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

No potential failure mechanism was identified and CIV WD-AOV-1728 would not be expected to fail and its safety function (to close on demand) would be expected to be performed based on the valve's acceptable test history.

There were no actual safety consequences for the event because there were no events requiring containment isolation and no releases via the containment sump pump discharge. Review of the recordings of the integrator for containment sump daily flow showed no operation for July 27. CIV WD-AOV-1728 was available to close on a Phase A containment isolation signal and terminate the release pathway.

In addition, the CIV has control room position indication that would allow operators to detect the failed CIV. Any releases as a result of a failed CIV would be detected. An open CIV and the open end of IVSWS valve IV-1646 would allow release into the PAB which contains a ventilation system (fans, roughing, HEPA, and charcoal filters) that exhausts PAB areas to the plant vent, a monitored release point.

There were no significant potential safety consequences of this event. The potential safety consequences of this event were considered under reasonable and credible alternative conditions. The plant is evaluated for a design basis accident (DBA) Loss of Coolant Accident (LOCA) and an assumed random single failure. The consequences of this event and a DBA with assumed single failure was assessed. The Indian Point 3 Individual Plant Evaluation (IPE) provides a Large Early Release Frequency (LERF) estimation of 7.53E-7 per year. The IPE assumes the highest bounding case of an interfacing LOCA or Steam Generator Tube Rupture initiator with no operator action to close a failed open CIV upon a Phase A actuation. Since the IPE defines a large release occurring only in lines greater than two inches in diameter, the postulated release through the 3/8 inch IVSWS valve IV-1646 is considered small under IPE assessments.

To quantify the events contribution, a probabilistic risk screening evaluation was performed with the conservative assumption that all Loss of Coolant Accidents (LOCAs) resulting in core damage produce a large early release through an assumed IVSWS valve opening of 3/8 inches. The assumptions are conservative since the actual IVSWS tubing inside diameter is 0.245 inches and not all core damage LOCA events result in large early releases. The contribution of all LOCAs resulting in core damage (considering all break sizes) is 8.89E-6 per year or 1.01E-9 per hour. Assuming an event duration of four hours, the conditional probability is 4.04E-9 which is risk insignificant. Although Indian Point 3 does not currently have the Standard Technical Specifications (STS), as a comparison to accepted risk, the STS contains an allowed outage time of four (4) hours for this condition. The potential dose consequences of a failure of CIV WD-AOV-1728 to close on a Phase A isolation during this event and a postulated DBA LOCA could have resulted in exceeding the dose limits of 10 CFR 50 Appendix A GDC-19 for the control room, and the offsite 10 CFR 100 dose limits for thyroid. The offsite whole body dose would be expected to remain within the limits of 10 CFR 100 based on previous assessments in LER 99-002 for a one inch opening.