**Indian Point 3**
**Nuclear Power Plant**
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**New York Power Authority**

**Robert J. Barrett**
Site Executive Officer

February 19, 1999
IPN-99-022

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 # 1999-002-00
**"Plant Outside Design Basis Due to Error In Original Design**
**Allowing Single Failure to Cause Loss of Containment Isolation"**

Dear Sir:

The attached Licensee Event Report (LER) 1999-002-00 is hereby submitted as required by 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73 (a)(2)(ii)(B) for a condition that was outside the design basis of the plant.

The Authority is making no new commitments in the LER.

Very truly yours,

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

cc: See next page

IE22

cc: Mr. Hubert J. Miller
Regional Administrator
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U.S. Nuclear Regulatory Commission
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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 | 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)



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**TITLE (4)**
Plant Outside Design Basis Due to Error In Original Design Allowing Single Failure to Cause Loss of Containment Isolation

| 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 |
| | | | | | | | | | N/A | 05000 |
| 01 | 22 | 1999 | 1999 -- | 002 -- | 00 | 02 | 19 | 1999 | N/A | 05000 |

| OPERATING MODE (9) | NA |
|---|---|
| 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) |
|---|---|
| Brian Young, Senior Mechanical Design Engineer | (914) 788-2361 |

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 January 22, 1999, with reactor power at approximately 100 percent, NYPA notified the NRC that a design condition that had the potential to place the plant outside its design basis was confirmed to exist. Operations declared the Vapor Containment Pressure Relief System inoperable. It was determined that considering a single failure of containment isolation valve VS-PCV-1190 to close upon demand, a potential containment release path (PS-SOV-1280) would exist. Use of the system was prohibited unless a one-hour Containment Integrity Limiting Condition of Operation action statement was entered, with subsequent action to close a manual isolation valve in the Weld Channel and Containment Penetration System supply/containment atmosphere exhaust line. The most likely cause of the event is an oversight during the initial design phase of the plant when considering single failure. Corrective actions included applying administrative controls to manually isolate the exhaust port of Weld Channel and Containment Penetration Pressurization System (WCCPPS) valve PS-SOV-1280 during pressure reliefs and installation of a manual valve. This event had no actual effect on the health and safety of the public.

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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 January 22, 1999, with reactor power at approximately 100 percent, a design condition that had the potential to place the plant outside its design basis was confirmed to exist. It was determined that considering a single failure of containment isolation valve {ISV} VS-PCV-1190 to close upon demand, a potential containment {NH} release path would exist. This issue was raised by the NRC Senior Resident Inspector to a System Engineer.

The System Engineer recorded the existence of a condition outside the plant's design basis in a Deviation Event Report (DER 99-00130) and corrective action was initiated to administratively restrict routine containment pressure relief. A one-hour non-emergency notification was made to the NRC (ENS Log #35298). Operations declared the Vapor Containment (VC) Pressure Relief System {VA} inoperable. Use of the system was prohibited unless a one-hour Containment Integrity Limiting Condition of Operation (LCO) action statement was entered, with subsequent action to close a manual isolation valve {V} in the Weld Channel and Containment Penetration Pressurization System (WCCPPS) {BD} supply/containment atmosphere exhaust line.

The WCCPPS supplies pressurized air to the spaces between the three in-series containment isolation valves (VS-PCV-1190, -1191 and -1192) in the VC Pressure Relief System. This air is supplied when the isolation valves are closed; closure is determined by the closed position limit switches of the isolation valves being "made". Removal (i.e., exhaust) of WCCPPS occurs when the isolation valve control switches are actuated to initiate operation of the VC Pressure Relief system. Supply of WCCPPS air is directed to each of the two isolation valve interspaces when the associated three-way solenoid valve {PSV} (PS-SOV-1280) is de-energized; a 1 inch line connects each solenoid valve to its corresponding isolation valve interspace. Exhaust of the WCCPPS air occurs through the same 1-inch line back through the solenoid valve's exhaust port to the Primary Auxiliary Building (PAB) {NF} atmosphere. This alignment and flow path occur when the solenoid valve is energized.

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A review of valve VS-PCV-1190's history was performed. The results from surveillance test 3PT-Q28, "Containment Isolation Valves PCV-1190, PCV-1191, and 1192 Pressure Relief System", were reviewed. The results from March 1982 to December 1998 indicated that valve PCV-1190 had never failed to meet the valve closing stroke time operability requirements of the Inservice Test Program. A sample review of pressure relief records from 1992 to 1998 did not identify any instances where VS-PCV-1190 failed to close after a containment pressure relief.

## CAUSE OF THE EVENT

The most likely cause of the event is an oversight during the initial design phase of the plant when considering single failure. The original design of the VC Pressure Relief System apparently failed to recognize that a single failure of the inboard containment isolation valve (VS-PCV-1190) to close during containment pressure relief, could prevent the isolation of the vent path from containment into the PAB during a postulated event.

## CORRECTIVE ACTIONS

The following corrective actions have been taken under the Authority's Corrective Action Program to address the cause of the event and to prevent it from re-occurring:

Caution tags were placed on the containment isolation valve controls to prevent opening of the valves without entering a one-hour LCO action statement.

The operating procedure for containment pressure relief (SOP-CB-3) was revised to address the new tagging requirements, closing the manual WCCPPS isolation valve, entering into a 1-hour LCO action statement for Containment Integrity and a 7-day LCO action statement for Weld Channel, and lifting a lead to PS-SOV-1280 to return the WCCPPS valve to its safe position while a Containment Pressure Relief was in operation.

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A temporary modification was installed on February 3, 1999, allowing the exhaust port of solenoid valve PS-SOV-1280 to be closed off following the venting of the interspace between the containment isolation valves, maintaining containment integrity when the inboard containment isolation valve (VS-PCV-1190) is open. Operating procedure SOP-CB-3 was revised to incorporate the requirements of the temporary modification and delete the requirement to lift the lead to the solenoid valve and enter a 7-day LCO for WCCPPS. Permanent modifications to the system are being evaluated.

Similar design errors in the future are expected to be precluded as a result of the development and evolution of the Design Change/Modification control process for Indian Point 3 (IP3). The Modification Control Process was introduced in about the 1988/1989 time frame and has been continuously improved over the years. The current process includes a series of guidelines and checklists which identify all system interfaces and design requirements, including failure effects and common mode failures.

An extent-of-condition review was performed of other containment isolation valves. The purpose of the review was to identify any other containment penetrations which might have similar configurations and which might provide a similar vent path outside containment. Two other containment penetrations, the Containment Building Purge Supply and Exhaust lines, were found to have similar configurations with regard to the WCCPPS interface, control logic and potential vent path. Currently, no further action is believed to be required for these valves based on their use being limited to cold shutdown. This is undergoing further evaluation.

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## ANALYSIS OF THE EVENT

The event is reportable under 10 CFR 50.73 (a) (2) (ii) (B). The Licensee shall report any condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

This event meets the reporting criteria because it was determined on January 22, 1999 that the containment isolation system can not meet its function of isolating Containment Building Pressure Relief line No. 68 during pressure relief, given a single failure of VS-PCV-1190 to close while pressure relief is in progress. This condition existed since original plant design and power operation (1975).

A review of the past two years of Licensee Event Report (LER) for events that involved system requirements not fully assessed during original design identified the following: LER 97-031, 97-020, and 97-006.

## SAFETY SIGNIFICANCE

This event had no actual effect on the health and safety of the public. There were no actual safety consequences for the event because there have been no events (Loss Of Coolant Accidents) requiring containment isolation.

A review of operating and testing experience dating back to 1982 has not identified any failures of VS-PCV-1190 to close upon demand.

The vent port of PS-SOV-1280 exhausts into the PAB. The PAB ventilation system ensures that this effluent would exhaust through the plant vent, a monitored release point.

The potential effect on the health and safety of the public was determined to be risk insignificant from a probabilistic viewpoint, and from a deterministic viewpoint valve VS-PCV-1190 would not be expected to fail to close during an event based on its past history.

Probabilistic Assessment

The potential safety consequences of this event were considered under postulated Design Basis Loss of Coolant Accident (LOCA) conditions. From the Indian Point 3 Individual Plant Evaluation (IPE), the Large Early Release Frequency (LERF) is estimated at 7.53E-7 per year. It assumes the highest bounding case of an interfacing Loss of Coolant Accident or Steam Generator Tube Rupture initiator with no operator action to close valve VS-PCV-1190 upon a Containment Isolation Phase A actuation. Since the IPE defines a large release occurring only in lines greater than 2" in diameter, the release through the 1" line 814 is considered small. To quantify its contribution, a probabilistic evaluation was then made assuming a Large-Break LOCA along with random failures of the containment isolation signal, valve hardware, and the interlocks between valves VS-PCV-1190 and PS-SOV-1280. A simple fault tree was constructed based on the operating circuit with component failure data taken from the IP3 IPE. A containment bypass (having both valves VS-PCV-1190 and PS-SOV-1280 open) during the large-break LOCA was used as the top event. The computed frequency is 1.14E-6 per year which is risk insignificant. For core damage to occur, additional random failures would have to be factored in and this would lower the frequency further.

In addition to the probabilistic assessment, the event was reviewed deterministically.

Deterministic Assessment

An evaluation of site boundary and low population zone radiation exposure dose consequences for a 1" diameter vent path from containment to the PAB following a design basis Large-Break LOCA was done and the results were compared with 10 CFR 100 limits. This was completed considering a postulated weld channel system vent path following pressure relief evolutions, valve VS-PCV-1190 failing to close, and a concurrent LOCA. The purpose of this evaluation was to specifically evaluate the consequence of this specific postulated event.

Final Safety Analysis Report (FSAR) assumptions were used and adjusted using ICRP-30 and expected PAB filtration after 2 hours.

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Radiation exposures at the Site Boundary (SB) and Low Population Zone (LPZ) were estimated assuming valve VS-PCV-1190 failed to close. The whole body radiation exposure results meet the 10 CFR 100 limit of less than 25 rem whole body. The results estimated for thyroid exposure exceed the 10 CFR 100 limit. The doses for the control room would exceed General Design Criteria, GDC-19 limits, however, it is not expected that these doses would affect the ability of an operator to respond to an event. The emergency plan has provisions for operators to don respiratory protective equipment, if necessary, for thyroid dose mitigation. Also, based on the past history of testing and operation of valve VS-PCV-1190, we believe it reasonable to assume that this valve would not fail to close upon demand.