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Nuclear Power Plant
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NewYorkPower Authority

Mr. Fred R. Dacimo
Plant Manager

June 19 2000
IPN-00-048

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 # 2000-005-00
Pressurizer Safety Valves Discovered Outside Their As-Found Lift Setpoint Test Acceptance Criteria After Removal and Testing; A Condition Prohibited by Technical Specifications

Dear Sir:

The attached Licensee Event Report (LER) 2000-005-00 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) for a condition recorded in the New York Power Authority's (NYPA) corrective action process as Deviation Event Report DER 00-01201.

NYPA is making no new commitments in this LER.

Very ~~truly~~ yours,

Fred R. Dacimo
Plant Manager
Indian Point 3 Nuclear Power Plant

cc: See next page

IE22

RGH-001

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)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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TITLE (4)

Pressurizer Safety Valves Discovered Outside Their As-Found Lift Setpoint Test Acceptance Criteria After Removal and Testing; A Condition Prohibited by Technical Specification

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
05	18	2000	2000	-- 005 --	00	06	19	2000		05000
										05000

OPERATING MODE (9): N

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)
Tom Orlando, Supervisor Performance & Reliability	(914) 736-8340

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
X	AB	RV	C710	YES					

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 May 18, 2000, performance engineering was notified by Wyle Laboratories that all three (3) pressurizer code safety valves (SV) were outside their As-Found (AF) lift setpoint test acceptance criteria (2461 - 2509 psig). The SVs were removed during the last refueling outage in the fall of 1999, and sent off site for testing. Wyle Labs used test 3PT-R005A which contains an AF setpoint pressure criteria of 2485 psig +/-1%. The SVs tested low but above the setpoint of the pressurizer Power Operated Relief Valves (2335 psig). Technical Specification (TS) 3.1.A.2.b requires that all pressurizer Code SVs be operable whenever the reactor is above cold shutdown. TS 3.1.A.2.c requires the SVs to be set at 2485 psig with +/-1% allowance for error but there is no AF setpoint specification. Discrepancies found in TS surveillance tests are assumed to occur at the time of the test unless there is firm evidence to believe that it occurred earlier. In accordance with NRC guidelines, the existence of similar discrepancies in multiple valves is an indication that the discrepancies may have arisen over a period of time and therefore the condition may have existed during plant operation and is therefore reportable. The cause of the condition was attributed to setpoint drift over the last operating cycle. The valves will be overhauled and parts replaced as necessary and As-Left (AL) tested to the correct setpoint prior to installation. The test will be revised to reflect the ASME Code AF allowable of +/-3%. The SV currently installed were AL tested to +/-1%. The event had no effect on public health and safety.

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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 May 18, 2000, at approximately 1300 hours, while at 100% steady state reactor power, Performance Engineering (PE) was notified by Wyle Laboratories that all three (3) pressurizer {AB} code safety valves (SV) {RV} were outside their As-Found (AF) lift setpoint test acceptance criteria (2461 - 2509 psig) after testing. The SVs were removed during the last refueling outage (RO10) in the fall of 1999, and sent off site to Wyle Laboratories for testing. Wyle Labs used surveillance test 3PT-R005A which contains an AF setpoint test pressure criteria of 2485 psig +/-1%. The test results had AF initial actuation pressure values of 2381 psig for SV RC-PCV-464, 2443 psig for SV RC-PCV-466, and 2430.5 psig for SV RC-PCV-468, all of which were below the test lower setpoint limit of 2461 psig. Technical Specification (TS) 3.1.A.2.b requires that all pressurizer Code SVs be operable whenever the reactor is above cold shutdown. TS 3.1.A.2.c requires the SVs to be set at 2485 psig with +/-1% allowance for error, but provides no AF setpoint specification. Discrepancies found in TS surveillance tests are assumed to occur at the time of the test unless there is firm evidence to believe that it occurred earlier. In accordance with NRC guidelines in NUREG-1022, the existence of similar discrepancies in multiple valves is an indication that the discrepancies may have arisen over a period of time, and therefore the condition may have existed during plant operation and is therefore reportable. PE assessed the condition, initiated a Deviation Event Report (DER-00-01201) and notified operations on May 19, at 0949 hours.

The three (3) pressurizer code SVs (RC-PCV-464, 466, 468) and the two (2) pressurizer Power Operated Relief Valves (PORV) {RV} (RC-PCV-455C, RC-PCV-456) protect the Reactor Coolant System (RCS) {AB} from overpressure during abnormal operating pressure and temperature conditions in accordance with the ASME B&PV code. The pressurizer code SV's are manufactured by Crosby Valve Division {C710}, Type/Model No. HB-BP-86.

CAUSE OF EVENT

The cause of the condition was attributed to random setpoint drift during the operating cycle, but SV overhaul and assessment is still pending.

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 pressurizer SVs (RC-PCV-464, 466, and 468) that were removed and AF tested will be overhauled and parts replaced as deemed appropriate and As-Left (AL) tested to the test acceptance criteria prior to installation. In accordance with the ASME Code, the valve that failed the +/-3% AF allowance will be refurbished and components replaced as required, the cause of the failure determined and a successful retest performed before being returned to service.

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

- TS 3.1.A.2 will be changed when the current TS are revised to the improved standard TS (ITS). The ITS will specify that surveillance testing of the SVs be performed in accordance with the Inservice Testing (IST) Program. The IST Program will test the SVs to ASME Section XI, OM-1 which has an AF criteria of valve nameplate pressure +/-3%. The ITS will prevent requiring an LER when AF testing of the SVs exceeds 2485 psig +/-1%, but meets the IST referenced acceptance criteria of 2485 psig +/-3%. Changes to the TS are awaiting NRC approval and implementation of the ITS which is scheduled for March 2001. Surveillance test 3PT-R005A will be revised to the AF valve pressure allowable range of 2485 psig +/-3% as an implementing action for the ITS.

ANALYSIS OF EVENT

The event is reportable under 10 CFR 50.73 (a) (2) (i) (B). The licensee shall report any operation or condition prohibited by the plant's Technical Specifications.

This event meets the reporting criteria because all three pressurizer code SVs failed to meet the AF test acceptance criteria. In accordance with NUREG-1022 Section 3.2.2 example 5, discrepancies found in TS surveillance tests are assumed to occur at the time of the test unless there is firm evidence to believe that the discrepancy occurred earlier. The NRC guidance further states that the existence of similar discrepancies in multiple valves is an indication that the discrepancies may have arisen over a period of time and therefore may have existed during operation and is therefore reportable. TS 3.1.A.2.b requires that all pressurizer Code SVs be operable whenever the reactor is above cold shutdown. With the SVs outside their test acceptance criteria, their test operability criteria was not met and per the NUREG-1022 guidance the condition existed during operation which is a condition prohibited by TS 3.1.A.2.b. Because there was at least one shutdown to the cold shutdown condition during the operating cycle, there was also a condition prohibited by TS 3.1.A.2.a, which specifies at least one SV shall be operable whenever the reactor head is on the vessel.

These SV were AL tested to the correct TS setpoint prior to return to operation from RO9 in the spring of 1997. When these SVs were tested after being removed after the RO10 outage, the AF results showed the SVs with setpoints outside the +/-1% test criteria but above the setpoint of the PORVs (2335 psig). The event would not be a TS prohibited condition under the proposed Indian Point 3 Improved Standard TS (ITS) which are under NRC review and scheduled to be implemented in March of 2001. The ITS has an AF SV setpoint that specifies the ASME/ANSI OM-1 allowed AF limit of 2485 +/-3%. Since only one SV was found outside the ITS limit, the condition would not meet the multiple SV test failure criteria of NUREG-1022 and therefore would not be reportable. The change to the ITS would have negated the need for reporting this event. A review of Licensee Event Reports (LER) covering the previous two SV operating cycles [RO7 (December 1990) through RO9 (May 1997)] did not identify any LERs reporting TS prohibited condition due to SV test failures.

SAFETY SIGNIFICANCE

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

There were no actual safety consequences for the event because there were no events per the safety analysis that required pressure relief of the RCS by the SVs during the time the SVs were installed. LER 1999-010-00 reported a PORV lift due to a pressure transient from a turbine runback and reactor trip. One PORV was unable to automatically lift due to loss of instrument control power. No SV lifted in response to this event.

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

There were no significant potential safety consequences of the event under reasonable and credible alternative conditions. Had the SV setpoint AF condition existed and a design basis accident or postulated transient occurred the condition would not have affected accident mitigation capability and the SVs overpressure function would have been met.

The SVs were discovered to have AF lift setpoints that drifted low therefore the required overpressure relief would have been performed. The impact of low SV setpoints on analyzed events were evaluated and determined to have no significant effect. The SV surveillance test has an AF acceptance criteria of +/-1%, but the ASME Code AF allowable is +/-3%, which two of the three SVs met. The current safety analysis assumes a SV lift setpoint tolerance band of +/-4%. Pressurizer relief at lower pressures impact departure from nucleate boiling ratio (DNBR), but the PORVs are set below the SVs and assumed to be operable for these analyzed events. The worst analyzed case for DNBR is the loss of external load event. This analysis with an assumed -4% SV setpoint tolerance results in a DNBR of 2.266, which is well above the analysis criteria DNBR limit of $\geq$ 1.54. The reactor protection system {JC} is designed to automatically terminate the analyzed transient before the DNBR falls below the limit value. Engineering judged that the AF test setpoint for SV RC-PCV-464 (2381 psig) which exceeded the analysis assumed tolerance of +/-4% (2400 psia/2385 psig) by less than 1%, was acceptable since the analysis has enough DNBR margin to accommodate an additional 1 to 2 % decrease from the assumed +/-4% setpoint tolerance. Therefore the minimum DNBR would be met for analyzed transients with the SV AF lift setpoint of 2381 psig.

Review of this event against the guidelines of NEI 99-02 Rev. 0, "Regulatory Assessment Performance Indicator Guideline," concluded it was not a safety system functional failure (SSFF) for the functional area of Primary System Safety and Relief. Although the pressurizer SV were considered inoperable, the safety function of RCS pressure relief could have been performed. Although the current surveillance test has a conservative setpoint of +/-1%, the AF setpoint limits allowed by ANSI/ASME Section XI OM-1 are +/-3% indicating the setpoint drift for two of three SVs were within code allowables. The setpoint drift was on the low side of the SV TS set pressure and above the PORV setpoint pressure therefore their safety function of RCS overpressure protection would have been performed. Also, per NUREG-1022 it is not necessary to consider a random single failure, absent an identified potential failure mechanism. Since no potential failure mechanism was identified and the PORVs and SVs would be expected to perform their safety function, no SSFF existed.