James A. FitzPatrick
Nuclear Power Plant
P.O. Box 41
Lycoming, New York 13093
315 342-3840

New York Power Authority

Harry P. Salmon, Jr.
Resident Manager

March 10, 1994
JAFP-94-0144

United States Nuclear Regulatory Commission
Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: LER-94-002:

Reactor Safety Relief Valve Setpoint Drift

Dear Sir:

This report is submitted in accordance with 10CFR50.73(a)(2)(i).

Questions concerning this report may be addressed to Mr. Donald Simpson at (315) 349-6361.

Very truly yours,

HARRY P. SALMON, JR.

HPS:DES:tlc

Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center

220023

IE22

NRC FORM 366 (5-92) 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 5/31/95

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)
James A. FitzPatrick Nuclear Power Plant	05000333	01 OF 04

TITLE (4)
Reactor Safety Relief Valve Setpoint Drift

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
02	22	94	94	002	00	03	10	94		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)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	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 (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	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)
Mr. Donald Simpson, Senior Licensing Engineer	(315) 349-6361

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
B	AD	T020	Y						

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)

During the Fall, 1993, maintenance outage, the pilot assemblies for two safety relief valves were removed for testing and recertification. On 2/22/94 and 2/23/94 the Authority received notification from the test facility that the test results exceeded the 1 percent tolerance allowed by Technical Specifications for valve actuation. Setpoint drift was a maximum of 1.8 percent on one valve and 2.5 percent on the second valve. A plant specific analysis performed previously envelopes the as-found setpoints. This analysis determined that setpoint drift greater than that found would have no significant safety impact on vessel overpressure margin, thermal limits, or Emergency Core Cooling system performance. Corrective action included replacing the SRV pilot assemblies with recertified assemblies and continued participation in the BWR Owners Group addressing setpoint drift.

LER numbers 92-016, 90-018, 89-026, 88-010, 88-004, 87-004, 85-013 and 85-009 are similar events involving SRV setpoint drift.

EIIS Codes are in []

Event Description

In November, 1993, during the conduct of a maintenance outage, the actuating mechanisms (pilots) from two safety relief valves (SRVs) [AD] were removed and sent to a test facility for testing, refurbishment and recertification. On February 23, 1994, the Authority received facsimile notification that one of the pilot mechanisms had actuated outside the 1 percent setpoint tolerance that is required by Technical Specification 2.2.1.B. On February 24, 1994, the Authority received facsimile notification that the second pilot mechanism also actuated outside the 1 percent setpoint tolerance. The initial set pressure observed for the SRV pilots were:

Plant Valve No.	Pilot Assembly Serial No.	Nameplate Set Pressure (PSIG)	Observed Initial Set Pressure (PSIG)	Deviation From Nameplate Percentage
02RV-71E	1056	1105	1113	0.7
02RV-71K	1047	1090	1117	2.5

Each pilot mechanism is tested four times. Although the 02RV-71E mechanism tolerance was acceptable on the initial test, set pressure was low by as much as 1.8 percent on the subsequent three test runs. The 02RV-71K mechanism set pressure was high by 2.5 percent on its initial lift then was observed to be low by as much as 2.2 percent on the subsequent test runs.

Cause

The pilot mechanisms are tested then disassembled, inspected and repaired (as needed) prior to recertification. Both pilot mechanisms had setpoints within 2.5 percent of nameplate. For this valve design, deviations of this magnitude from nameplate set pressure are not unusual. Because the initial lift setpoint drift was high for both pilot mechanisms, it is possible that there was some pilot disc to seat corrosion induced bonding. This observation is supported by subsequent test runs which showed uniformly lower lift setpoints.

Both pilot mechanisms required refurbishment because of leakage observed at normal operating pressure both before and after the test runs.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95

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.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	94	002	00	03 OF 04

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

Analysis

The observed setpoint of each SRV pilot mechanism deviated by more than 1 percent from the values specified in Technical Specification 2.2.1.B. Therefore, this occurrence is reported under the provision of 10CFR50.73(a)(2)(i)(B) as an operation of the plant in a condition prohibited by the Technical Specification. The remote actuation (operator demand) and automatic depressurization system (ADS) functions would not have been effected by this event. An analysis to determine the effects of SRV setpoint drift was initiated as a result of earlier similar events (LER-87-004 and LER-88-004) and has been completed.

This analysis considered plant operation with two ADS SRVs inoperable and established an upper bound for the remainder of the SRVs. The analysis showed that continuous operation of the plant would be acceptable with nine SRVs actuating at 1195 psig. The acceptance criteria for this analysis was a 50 psi margin to the ASME code upset reactor vessel pressure limit of 1375 psig during the limiting overpressure event. Additionally, the analysis confirmed that setpoint drift of nine SRVs to the 1195 psig limit would not adversely affect the following:

- High Pressure Coolant Injection (HPCI) [BJ] system
- Reactor Core Isolation Cooling (RCIC) [BN] system
- Primary Containment [NH] integrity
- Fuel Thermal Limits
- Emergency Core cooling System (ECCS)/Loss of Coolant Accident (LOCA) performance

This analysis bounds the SRV lift setpoints identified in this event since neither SRV exhibited a setpoint approaching 1195 psig.

Based on the bounding evaluation, it is concluded that the setpoint drift of the two pilot mechanisms did not represent any hazard. Plant response to any of the accident conditions described in the Final Safety Analysis Report (FSAR) would have been acceptable. Therefore, the safety significance of this occurrence is low.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	94	002	00	04 OF 04

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

Corrective Actions

1. The pilot assemblies were replaced with refurbished and recertified assemblies prior to startup following the maintenance outage. The removed pilot assemblies will be refurbished and recertified for future installation.

2. All SRVs, rather than half as specified in the Technical Specifications, will continue to be subjected to test, refurbishment and recertification once each operating cycle.

3. The Authority will continue its participation in the BWR owners group to address the SRV setpoint drift issue.

Additional Information

Failed Component Identification: Manufacturer: Target Rock Corp
Model Number: 7567F-010
NPRDS Manufacturer Code: T020
NPRDS Component Code: Valve

Similar Events: LER-85-009, 85-013, 87-004, 88-004, 88-010, 89-026, 90-018 and 92-016, are similar events which reported SRV setpoint drift.