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New York Power Authority

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

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 92-016-00
REACTOR SAFETY RELIEF VALVE SETPOINT DRIFT

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(i).

Question concerning this report may be addressed to Mr. Gerald Ottman at (315) 349-6548.

Very truly yours,

Harry P. Salmon

HARRY P. SALMON, JR.
RESIDENT MANAGER

HPS:GIO:llm

Enclosure

cc: USNRC, Region I
USNRC Resident Inspector
INPO Records Center
JAFP File
RMS - JAF

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

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TITLE (4)

REACTOR SAFETY RELIEF VALVE SETPOINT DRIFT

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 NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 3	2 6	9 2	9 2	0 1 6	0 0	0 4	1 6	9 2		0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10):

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (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: AREA CODE	
Gerald I. Ottman, Jr.	3 1 5	3 4 9 - 6 5 4 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	A D	R V	T 0 2 0	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 fifteen single-space typewritten lines) (16)

EIIS Codes are in [ ]

During the cu nt plant refueling outage, the pilot assemblies for all eleven safety relief valves [AD] were removed for testing and recertification. 'n 3/26/92 the Authority received notification from the test facility ..at eight of the eleven valves tested actuated at pressures which exceeded the 1% setpoint tolerance allowed by Technical Specifications. Setpoint drift varied from 1.2% to 8.6%.

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 includes replacing all eleven SRV pilot assemblies with recertified assemblies and continued participation in the BWR Owners Group addressing setpoint drift.

LER numbers 90-018, 89-026, 88-010, 88-004, 87-00., 85-013 and 85-009 are similar events involving SRV setpoint drift.

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DESCRIPTION

During the current refueling outage, which began January 11, 1992, the actuating mechanisms (pilots) from all eleven safety relief valves (SRVs) [AD] were removed and sent to a test facility for testing, refurbishment and recertification. On 3/26/92 the Authority received written notification that eight SRV pilots had actuated outside the 1% setpoint tolerance that is required by Technical Specification 2.2.1.B. The initial set pressure observed for the SRV pilots was:

Plant Valve No.	Pilot Assembly Serial No.	Nameplate Set Pressure (PSIG)	Observed Initial Set Pressure (PSIG)	Deviation From Nameplate psi	%
02RV-71B	1217	1140	1154	14	1.2
02RV-71C	1218	1140	1210	70	6.1
02RV-71D	1050	1105	1122	17	1.5
02RV-71E	1080	1105	1119	14	1.3
02RV-71F	1012	1140	1156	16	1.4
02RV-71H	1013	1140	1227	87	7.6
02RV-71K	1047	1090	1155	65	6.0
02RV-71L	1088	1090	1184	94	8.6

CAUSE

All pilots are disassembled, inspected and repaired (as needed) prior to recertification. Four of the valves (S/N 1012, 1050, 1080, 1217) had setpoints within 2% of nameplate. Inspection of these valves did not detect any condition which would cause a variation in the setpoint. For this valve design, deviations of this magnitude from nameplate set pressure are not unusual.

For the other four valves, which experienced setpoint drift between 6.0 and 8.6 percent of nameplate, the following causes have been determined:

- For pilot S/N 1013, 1047 and 1218 the initial test showed setpoint drift, however subsequent tests (3 additional per valve pilot) for all pilots were within 2.3% of nameplate setpoint. This is typical for pilots experiencing pilot disc to seat corrosion induced bonding. Examination of the pilots did not indicate any other problems which would contribute to the observed setpoint drift.

- For pilot S/N 1088, severe leakage at the pilot disc was experienced. Due to this leakage, the test facility was unable to obtain thermal equilibrium of the valve prior to test start and testing had to be terminated after two test runs. Examination at the test facility indicated severe steam cutting of the pilot disc and seat. Steam cutting to this degree can be expected to cause a variation in setpoint due to the reduction in area of the pilot and the fact that severe leakage allows slight pressurization of the area on top of the pilot disc. The second test run performed on this pilot also showed similar setpoint variation (7.2 percent above nameplate).

ANALYSIS

The observed setpoint of eight SRVs deviated by more than one percent from the values specified in Technical Specification 2.2.1.B. Therefore, this event 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 Specifications. 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 9 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 9 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 setpoints identified by testing since only two SRVs exhibited setpoints in excess of 1195 psig.

Based on the bounding evaluation, it is concluded that the setpoint drift of the valves 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.

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

CORRECTIVE ACTION

1. The pilot assemblies will be replaced with refurbished and recertified assemblies prior to startup following the current refuel outage. All 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 plant will continue to participate in the Boiling Water Reactor Owners Group (BWROG) SRV Setpoint Drift Fix Committee. This committee is currently completing a design modification to the SRV pilot to mitigate the corrosive bonding causing setpoint drift. SRVs with modified pilots will be installed in operating plants starting late in 1992 to verify the effectiveness of the modified design.

ADDITIONAL INFORMATION

Failed Component Identification:

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

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