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JAFP-90-0512

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

REFERENCE: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 89-026-01
Safety Relief Valve Setpoint Drift

Dear Sir:

Enclosed is Revision 1 to the Licensee Event Report which was originally submitted in accordance with 10 CFR 50.73(a)(2)(i) on January 26, 1990.

This supplemental report revises the "Description" and "Cause" sections to provide the results of additional setpoint testing and vendor examinations of the two safety relief valves which was received on June 15, 1990.

Questions concerning this report may be addressed to Mr. Hamilton Fish at (315) 349-6013.

Very truly yours,

WILLIAM FERNANDEZ

WF:HCF:lar

Enclosure

cc: USNRC, Region I
INPO Records Center
American Nuclear Insurers
NRC Resident Inspector

Cert No P229769383
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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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Reactor Safety Relief Valve Pilot Assembly 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)
1 2	2 6	8 9	8 9	0 2 6	0 1	07	0 5	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 1 0 0

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	
Hamilton C. Fish	3 1 5	3 4 9 - 6 0 1 3

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)	[X] NO				

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

UPDATE REPORT - PREVIOUS REPORT DATE 1/26/90
EIIS Codes are in []

Following a reactor scram on November 5, 1989, two safety relief valves (SRVs) [AD] were removed for testing. The valve test facility provided written notice, received by the Authority on December 20, 1989, that both valves actuated at setpoints which deviated from the design point by more than the +/-1% allowed by Technical Specifications. SRV "E" lifted early at -1.4%. SRV "F" lifted at +4.7%. Disassembly and examination of the pilot mechanisms found steam cuts on the pilot valve disc seat and bellows for SRV "F". Disc to seat bonding is believed to be the cause of the high initial lift pressure for SRV "F". No deterioration was noted on SRV "E" components and no cause for early lifting of that valve was determined.

Evaluation of reactor pressure relief capability shows operation would be acceptable with 2 of 11 SRVs inoperable and a setpoint tolerance of +/-3%. Corrective actions included replacing failed SRVs with recertified valves, continued participation in the BWR Owners' Group to resolve SRV issues, and submission to the NRC of proposed changes to Technical Specifications to take credit for excess installed SRV capacity.

LER-85-009, 85-013, 87-004, 88-004, and 88-010 are similar events involving SRV setpoint drift.

NRC Form 366 (9-83)

UPDATE REPORT - PREVIOUS REPORT DATE 1/26/90

EIIS Codes are in []

Description

A reactor scram from full power occurred at 3:23 P.M. on November 5, 1989 which was attributed to problems with the turbine electro-hydraulic control system [JJ] (LER-89-020). Peak reactor pressure reached 1082 psig during the turbine control valve closure transient which was below the relief design setpoint for any of the safety relief valves [AD].

Safety relief valve (SRV) "F" had a relief design setpoint of 1140 psig (58 psi above the transient pressure). The recorded increase in SRV exhaust pipe temperature for this valve indicates that it passed a quantity of steam above that normally associated with pilot valve leakage. However, the absence of alarms from the SRV tailpiece acoustic monitor combined with the instrumented exhaust pipe temperature rate of rise indicated that the valve did not actually lift. This was confirmed by the exhaust pipe temperature which was below that associated with the lifting of an SRV.

Safety relief valve "E" was mounted on the same line as SRV "F". The SRV "E" design relief setpoint of 1105 was 27 psi above the peak pressure recorded during the transient. SRV "E" performed as expected and did not lift. SRV "E" would have been expected to lift before SRV "F" if the pressure had actually risen to the SRV "F" setpoint of 1140 psig.

The activating topworks mechanisms were removed from both SRV "E" and "F" during the outage following the scram. The topworks were sent to a contract laboratory facility for testing and vendor examination.

Written notification from the contract laboratory received on December 20, 1989 informed the plant staff that the two SRVs which were tested did not actuate within +/-1% of the nameplate setpoint as required by Technical Specification 2.2.1.B. The initial set pressure observed by the contractor at his facility were:

Plant Valve Number	Pilot Assembly Serial No.	Nameplate Set Pressure (psig)	Observed Initial Set Pressure (psig)	Difference From Specification In PSI	%
02RV-71E	1050	1105	1089	-16	-1.4%
02RV-71F	1087	1140	1194	+54	+4.7%

Following the initial as-found lift (1,089 psig, -1.4%), SRV "E", S/N 1050, was tested three more times with these results:

Observed Lift Pressure	Deviation from Nameplate (1,105 psig) PSI	%	Technical Specification Limit (+/-1%)
1098	-7	-0.6	Acceptable
1095	-10	-0.9	Acceptable
1086	-19	-1.8	Exceeds Limit

Prior to the lift testing no pilot valve leakage was observed and the pilot disc was not stuck.

During the pretest heat-up of the SRV "F", S/N 1087, pilot assembly, the valve began leaking with 100 psig being applied to the valve inlet. The leakage was so severe that the test facility exhaust system was being overloaded. At the request of the test facility vendor, the Authority authorized immediate set pressure actuation without achievement of thermal equilibrium. Four set pressure actuations were run. The severe leakage continued throughout the test. The following data was obtained from the tests of valve 02-RV-71F, S/N 1087, which had a nameplate set pressure of 1140:

Run No.	Observed Set Pressure (psig)	Reseat Pressure (psig)	Percent Reseat	Delay (msec)	Remarks
1	1194	822	68.8	450	Leaking Severely
2	1108	821	73.5	310	Leaking Severely
3	1115	824	73.2	280	Leaking Severely
4	1108	821	74.0	280	Leaking Severely

Cause

A detailed examination of the valve pilot assemblies was performed by the vendor to determine possible causes during April 1990. All components of SRV "E", S/N 1050, were found to be satisfactory by the vendor. No cause was determined for the slightly early (1.4%) lifting of this SRV.

The test results of SRV "F" (in which severe leakage was demonstrated) are consistent with the indicated leakage of the valve following the scram. The high lift pressure of the first run followed by a series of lifts at a consistently lower setpoint is characteristic of a disc sticking problem. Vendor examination of SRV "F", S/N 1087, found steam cuts in the pilot disc seat and bellows. The high initial lift pressure (+54 psi), followed by low (early) lift pressures, is

indicative of bonding of the seat to the disc. This bonding is believed to be the cause for delayed lifting. Sticking of the pilot disc could not be confirmed by diagnostic testing due to the excessive steam leakage.

Analysis

The observed setpoint of two SRVs deviated by more than 1% from the values specified in Technical Specification 2.2.1.B. Therefore, this event is reported under the provision of 10 CFR 50.73(a)(2)(i)(C) as a deviation from the plant's Technical Specifications.

The remote actuation (operator demand) and automatic depressurization system (ADS) functions would not have been effected by this event. An evaluation 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.

The evaluation considered the following:

- relaxation of the +/-1% nominal valve nameplate setpoint tolerance to +/-3%, and

- operation with any 2 SRVs inoperable, and

- setting all 11 SRVs at a single nominal nameplate setpoint.

The results of a bounding evaluation show that continuous operation of the plant would be acceptable with a 50 psi margin to the American Society of Mechanical Engineers (ASME) Code upset reactor vessel pressure limit of 1375 psig during the limiting overpressure event with any 9 of the 11 SRVs operable and with a common valve actuation pressure of 1195 psig. This analysis bounds the conditions identified by the SRV testing.

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.

Corrective Action

Immediate Corrective Action: The valves were replaced with refurbished and recertified valves. The failed SRVs will be refurbished and recertified for future installation. Although there was no indication of a failure of SRV "E", it was replaced as a conservative prudent action because it was mounted on the same line as SRV "F" which was leaking.

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

Long-Term Corrective Action:

1) A proposed change to Technical Specifications to allow continuous operation of the plant with SRV operations consistent with the new analysis was submitted to the NRC on December 20, 1989.

2) The plant has previously modified SRVs by installing pilot valve discs made of a different material. This action is part of the Boiling Water Reactor Owners' Group (BWROG) plan for correction of the Target Rock 2-Stage SRV drift problem. The plant has previously installed and operated with approximately half of the valves containing the new pilot valve discs made of PH13-8Mo. Testing of these valves is scheduled during the spring 1990 refueling outage. Further modifications will be dependent on the results of this testing and the evaluation being performed by the BWROG SRV Setpoint Drift Fix Committee.

3) All SRVs (rather than half of the valves as specified by Technical Specifications) will continue to be subjected to test, inspection, refurbishment, and recertification once each operating cycle until the test data, or the BWROG SRV Setpoint Drift Fix Committee recommendations indicate otherwise, or the Technical Specifications are amended.

Additional Information

Failed Component Identification:

- SRV Manufacturer: Target Rock Corp.
- Valve Model Number: 7567F
- Manufacturer NPRD Code: T020

LER-85-009, 85-013, 87-004, 88-004, and 88-010 are similar events which reported SRV setpoint drift. LER-89-020 describes the scram which initiated testing of the SRVs reported in this LER.