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)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
James A. FitzPatrick Nuclear Power Plant	05000333	01 OF 06

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 NAME	DOCKET NUMBER 05000
03	15	95	95	006	01	08	29	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	0

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
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
	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. Gordon Brownell, Senior Licensing Engineer	(315) 349-6360

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

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 1994/1995 refueling outage, all eleven safety relief valve (SRV) pilot assemblies were removed for testing and recertification and ten exceeded the 1 percent valve actuation tolerance allowed by Technical Specifications. SRV testing results for five SRVs were reported in LER-95-001. On 3/15/95, the Authority received notification from the test facility that five additional SRVs exceeded the 1 percent tolerance with a set point drift ranging from -1.56 percent to 10.86 percent. A previous plant specific analysis determined that operation of the plant would be acceptable with nine SRVs actuating at 1195 psig. Two SRVs in this report and two in the previous report actuated at greater than 1195 psig. Consequently, a new analysis was completed using the as-found actuation pressures of these eleven SRVs to determine the impact on the margin to the reactor vessel pressure safety limit. The analysis concluded that, although there was a reduction in this margin, the higher SRV actuation pressures did not result in a violation of the ASME reactor vessel peak allowable pressure limit. Corrective actions included replacing SRV pilot disc material and participation in the BWR Owners Group SRV Committee. Similar LERs are 95-001, 94-002, 92-016, 90-018, 89-026, 88-010, 88-004, 87-004, 85-013, and 85-009.

Event Description

During the 1994/1995 refueling outage, all eleven safety relief valve (SRV) pilot assemblies were removed and sent to a test facility for testing, repair (as needed), and recertification. On January 6, 1995, the Authority was notified that five of six pilot valves had actuated outside the 1 percent setpoint tolerance required by Technical Specification 2.2.1.B. See LER-95-001. On March 15, 1995, the Authority was notified that the remaining five pilot valves had actuated outside the 1 percent setpoint tolerance required by Technical Specification 2.2.1.B. The initial actuation pressure and percent deviation from the set pressure for the ten SRV pilots was:

Plant Valve No.	Pilot Assembly Serial No.	Nameplate Set Pressure (psig)	Initial Actuated Pressure (psig)	Deviation From Nameplate Percentage
02RV-71A	1045	1140	1152	1.05%
02RV-71B	1110	1140	1167	2.37%
02RV-71C	1053	1140	1173	2.89%
02RV-71D	1080	1105	1225	10.86%
02RV-71E	1056	1105	1129	2.17%
02RV-71F	1087	1140	1153	1.14%
02RV-71G	1052	1140	1206	5.79%
02RV-71H	1111	1140	1197	5.00%
02RV-71K	1051	1090	1213	11.28%
02RV-71L	1047	1090	1073	-1.56%

Each pilot assembly is normally tested four times. The initial actuation is used to calculate Technical Specification tolerance. Nine of the ten pilot assemblies were found out of tolerance with upward set point drift. 02RV-71L was found out of tolerance with a downward set point drift and was the only one with a platinum alloy disc.

Cause

The most likely cause of SRV set point drift is corrosion bonding between the SRV pilot disc and pilot seat. This conclusion is based on metallurgical examinations of pilot discs from many affected SRVs (reference GE-NE-126-E031-0692, DRF B21-00491). This BWR Owners Group SRV Committee report states that radiologically produced hydrogen and oxygen can concentrate in the immediate vicinity of the pilot disc and seat interface (cavity) as a result of reactor steam condensation and attributes oxygen concentration in this area to be the most likely cause of corrosion bonding and therefore, the most likely contributor to SRV upward set point drift.

The initial and subsequent actuations and the approximate amount of time (service time) that steam was applied to each pilot assembly was:

Plant Valve No.	Set Pressure (psig)	Actuated Pressure (psig) 1st	2nd	3rd	4th	Service Time Months
02RV-71A	1140	1152	1146	1142	1138	22
02RV-71B	1140	1167	1142	1135	1136	22
02RV-71C	1140	1173	1135	1136	1135	22
02RV-71D	1105	1225	1128	1111	1124	22
02RV-71E	1105	1129	1098	Note 1		7
02RV-71F	1140	1153	1161	1153	1158	22
02RV-71G	1140	1206	1167	1176	1171	22
02RV-71H	1140	1197	1150	1141	1142	22
02RV-71K	1090	1213	1113	1105	1103	12
02RV-71L	1090	1073	1088	1088	1085	7

Note 1: Test 3 and 4 were not conducted for 02RV-71E due to seat leakage

The four pilot assemblies that were above three percent on the initial actuation pressure dropped to within three percent on subsequent lifts and therefore, it is likely that there was pilot disc to seat corrosion induced bonding for those four. This conclusion is consistent with GE-NE-126-E031-0692, DRF B21-00491. The four pilot assemblies referred to were in positions 02RV-71D, G, H. and K.

Analysis

The observed actuations for ten of eleven pilot assemblies deviated by more than 1 percent from the values specified in Technical Specification 2.2.1.B. Technical Specification Amendment 217, issued on September 28, 1994, and effective upon startup following the 1994/1995 refuel outage, modified SRV performance limits to provide a single nominal setpoint for all valves to 1110 psig, a setpoint tolerance of 3 percent, and to allow for two SRVs to be inoperable during continuous power operation. This occurrence is, however, being 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, NEDC-31697P-2, Revision 2, to determine the effects of SRV setpoint drift was initiated as a result of earlier similar events (LER-88-004 and LER-94-002) and has been completed. This analysis considered plant operation with two 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 the remaining nine SRVs actuating at 1195 psig. The acceptance criteria for this analysis was a 50 psi margin to the ASME Boiler and Pressure Vessel Code reactor overpressure protection limit of 1375 psig during a limiting overpressure event. Additionally, the analysis confirmed that setpoint drift of nine SRVs to the 1195 psig limit (with two SRVs inoperable) would not adversely affect the following:

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

A new analysis, GE-NE A00-05785-1, Revision 0, JAF Cycle 11 SRV Drift Analysis, was performed to evaluate the impact of the four SRVs with as-found actuation pressures higher than the 1195 psig calculated limit used in analysis NEDC-31697P-2, Revision 2. Conservative analysis calculations assumed that two of the eleven SRVs with the lowest as-found actuation pressures were out of service. This analysis concluded that the higher as-found SRV set points did not result in a violation of the ASME reactor vessel peak allowable pressure limit. However, the impact of these higher as-found set points reduced the margin to the overpressure protection limit of 1375 psig from 75 to 57 psi.

Analysis (cont)

The supporting facts listed below are relevant as they, in qualitative terms, illustrate a high degree of confidence that the SRVs will perform their design basis safety function during the 1995-1996 operating cycle:

1. The FitzPatrick Plant SRVs have realized a net increase in drift margin to the upper limit by implementing a single set point. In January 1995, all eleven SRVs were set to a lift set point of 1110 psig. This resulted in a net increase in drift margin (to 1195 psig) of 30 psig for the seven SRVs that were previously set to 1140 psig. This also resulted in an average reduction in drift margin of 12.5 psig for four SRVs that were previously set (two valves) at 1105 psig and (two valves) at 1090 psig.

2. The SRV configured with the Platinum alloy disc functioned as designed. It did not exhibit an upward set point drift during the 1993-1994 operating cycle. Four of the eleven SRVs have been configured with this disc material for the 1995-1996 operating cycle. It is reasonable to expect these four pilot assemblies to exhibit improved set point drift characteristics when tested at the end of this cycle.

3. In the 1993-1994 operating cycle, all of the SRVs would have lifted to relieve pressure although some at elevated lift pressures. This indicates that the degradation mechanism seen at the FitzPatrick Plant affects set points but does not preclude SRV lifting.

Corrective Actions

1. The pilot assemblies were replaced with refurbished and recertified assemblies prior to startup. 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. Four currently installed pilots (S/N's 1050, 1062, 1088 and 1217) have platinum alloy pilot discs which is the latest effort by the BWR Owner's Group that provides an alternate disc material in an effort to limit disc to seat bonding due to corrosion.

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TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
James A. FitzPatrick Nuclear Power Plant	05000333	95	006	01	06 OF 06

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

Corrective Actions (cont)

4. The Authority will continue its participation in the BWR Owners Group to address the SRV setpoint drift.

Additional Information

A. Failed Component

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

B. System and Component Identification:

SYSTEM - COMPONENT	IEEE803A	EIIS
Main Steam System	N/A	SB
Automatic Depressurization System	N/A	JC
High Pressure Coolant Injection System	N/A	BJ
Reactor Core Isolation Cooling System	N/A	BN
Primary Containment	N/A	NH
Safety Relief Valve	RV	N/A

C. Similar Events:

LER-95-001, 94-002, 92-016, 90-018, 89-026, 88-010, 88-004, 87-004, 85-013, and 85-009

D. References:

GE-NE-126-E031-0692, DRF B21-00491; Evaluation of the BWROG SRV Set Point Drift Fix Sub-Committee's Proposed Design Modification for the Target Rock Two-Stage Safety Relief Valve Design, Model No. 7567F

NEDC-31697P-2, revision 2; Updated SRV Performance Requirements for the JAFNPP.

GE-NE-A00-05785-1, DFR A00-05785; FitzPatrick Cycle 11 SRV Drift Analysis.