William S. Orser
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10CFR50.73

November 20, 1992
NRC-92-0131

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Licensee Event Report (LER) No. 92-009

Please find enclosed LER No. 92-009, dated November 20, 1992, for a reportable event that occurred on October 21, 1992. A copy of this LER is also being sent to the Regional Administrator, USNRC Region III.

If you have any questions, please contact Barb Siemasz, Compliance Engineer, at (313) 586-1683.

Sincerely,

WSOrser

Enclosure: NRC Forms 366, 366A

cc: T. G. Colburn
A. B. Davis
M. P. Phillips
W. J. Kropp
P. L. Torpey

Wayne County Emergency
Management Division

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4): Safety Relief Valves Set Pressure Outside Technical Specification Limit

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 0	2 1	9 2	9 2	0 0 5	0 0	1 1	2 0	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 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	
Barbara G. Siemasz, Compliance Engineer	3 1 3	5 8 6 - 1 6 8 3

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 fifteen single-space typewritten lines) (16)

The Main Steam System is equipped with fifteen Safety Relief Valves (SRVs). Technical Specification 4.4.2.1.2 requires that one half of the SRVs be proven operable at least once every eighteen months by performing a set pressure test. Fifteen SRV pilot assemblies were removed during the third refueling outage and sent to Wyle Laboratories to meet the surveillance requirement. Wyle Laboratories notified Detroit Edison that one valve failed to actuate during testing and eight other valves failed their set pressure test for a total of nine failures.

The cause of this event has been under review by Detroit Edison and the Boiling Water Reactor Owners Group (BWROG) SRV Setpoint Drift Fix Development Committee. The test data showed that the pilot valve, pilot disc-to-seat sticking was due to corrosion binding and is the primary cause for deviations from the set point tolerance range. An analysis was performed which shows that the peak dome pressure experienced during the event of a vessel over-pressurization would be within the safety limits.

The referenced SRV pilot assemblies have been replaced by refurbished and certified spare assemblies, which have been recertified to reflect the new power uprate setpoint ranges. The removed SRV pilot assemblies will be refurbished and recertified in preparation for installation during the fourth refueling outage. In addition, Detroit Edison is continuing to follow industry experience to resolve this issue.

NRC Form 366 (9-83)

Initial Plant Conditions:

Operational Condition: 5 (Refueling)
Reactor Power: 0%
Reactor Pressure: 0 psig
Reactor Temperature: 91° fahrenheit

Description of Event:

The Main Steam System (SB) is equipped with fifteen Target Rock two-stage pilot-operated Safety Relief Val es [(SRVs) (RV)], which are designed to prevent over-pressurization of the Nuclear Steam Supply System (NSSS). Technical Specification surveillance requirement 4.4.2.1.2 states that half of the SRVs must be proven operable at least once every eighteen months by performing a set pressure test. All fifteen SRVs must be set pressure tested during a forty month period.

Fifteen SRV base body/pilot assemblies were removed during the third refueling outage, which began on September 12, 1992, and sent to Wyle Laboratories to meet the surveillance requirement. On October 21, 1992, Wyle Laboratories notified Detroit Edison that nine SRVs exceeded their Technical Specification acceptance tolerance of ±1% of their pressure setpoint.

The SRVs are divided into three pressure setpoint groups. The first group consists of five valves set to open when Reactor Pressure Vessel (RPV) pressure equals or exceeds 1110 psig; the second group consists of five valves set to open when RPV pressure equals or exceeds 1120 psig; and the third group consists of five valves set to open when RPV equals or exceeds 1130 psig. Following is a table summarizing the results of the SRVs tested:

TABLE NO. 1 - SRV TESTING RESULTS

VALVE NUMBER	SET PRESSURE (psig)	MEASURED ACTUAL SETPOINT (psig)		ACTUAL % DRIFT
B2104-F013A	1110 ±11 psig	1123 psig		+ 1.17%
B2104-F013B(4)	1110 ±11 psig	1195 psig		+ 7.66%
B2104-F013G	1110 ±11 psig	1097 psig		- 1.18%
B2104-F013C(3)	1110 ±11 psig	1112 psig		+ 0.18%
B2104-F013K	1110 ±11 psig	1152 psig		+ 3.78%
B2104-F013N(3)	1120 ±11 psig	1110 psig		- 0.89%
B2104-F013M(3)	1120 ±11 psig	1110 psig		- 0.89%
B2104-F013L(3)	1120 ±11 psig	1125 psig		+ 0.45%
B2104-F013F	1120 ±11 psig	1130 psig	(1)	+ 0.89%
B2104-F013D(3)	1120 ±11 psig	1126 psig		+ 0.54%
B2104-F013R	1130 ±11 psig	1159 psig		+ 2.57%
B2104-F013P(4)	1130 ±11 psig	1212 psig		+ 7.25%
B2104-F013J	1130 ±11 psig	1148 psig		+ 1.59%
B2104-F013H	1130 ±11 psig	1174 psig		+ 3.89%
B2104-F013E	1130 ±11 psig	>1250 psig	(2)	>+ 10.62%

Notes:

(1) Recorder malfunctioned - recorded setpoint from visual observation of Heise pressure gauge.

(2) Test terminated - valve exceeded the design pressure of 1250 psig and would not actuate.

(3) Valves which met the T.S. 4.4.2.1.2 tolerance of ±1%.

(4) Pilot disc material ARMCO Alloy PH13-8MO Steel.

Nine of the fifteen SRVs failed to lift within their setpoint tolerance range during as-found testing conducted at Wyle Laboratories. Eight of nine failures were due to upward setpoint drift beyond +1% of setpoint; five of this eight had setpoint drift above 3% with one valve not lifting during the test sequence (setpoint greater than 1250 psig). One SRV test failure was due to downward drift with its setpoint found at -1.18%.

Cause of Event:

The cause(s) of this event has been under review by Detroit Edison and by the Boiling Water Reactor Owners Group (BWROG) SRV Setpoint Drift Fix Development Committee. A summary review of the cause(s) is presented in the text below.

The test data shows that pilot disc-to-seat sticking is the primary cause for deviations from the setpoint tolerance range. This problem has been evaluated by the BWROG and found to be caused by pilot disc and seat corrosion bonding. Corrosion bonding occurs because temperature of the pilot assembly is lower than the main steam system piping and, since the pilot assembly is mounted on top of the piping, it tends to act as a concentrating cell. Main steam oxygen content due to radiolytic decomposition of water in the reactor and feedwater entrained air is about 18 to 21 ppm. Estimated concentration levels in the pilot assembly due to the temperature drop are on the order of about 50,000 ppm O_2 or higher, based upon BWROG analyses. There has been no other contaminant of an elemental or molecular composition identified as a contributing factor to the corrosion. Additionally, there were no abnormal chemistry events during the Fermi 2 Cycle 3 operation. Corrosion bonding results in an increase in the differential pressure needed to lift the pilot disc, thus causing an upward drift in the SRV setpoint.

Analysis of Event:

Detroit Edison has analyzed several cases to ascertain the sensitivity of vessel peak pressure to changes in SRV actuation pressure. The results of this analysis for Cycle 3 operation is presented below:

SUMMARY OF PEAK VESSEL PRESSURE RESULTS

CASE	PEAK DOME PRESSURE (psia)
Case 1, 11 Operable SRVs	1279
Case 2, 14 Operable SRVs	1263
Case 3, 11 Operable SRVs	1285

Case 1: Baseline calculation for Cycle 3 assumes 11 operable SRVs are capable of lifting within 1% of rated set pressure with the 4 inoperable valves at the lowest set pressure.

Case 2: Calculation performed assuming 14 operable SRVs; 8 with set pressures of 1131 psig, 3 with set pressures of 1160 psig and 3 with set pressures of 1212 psig.

Case 3: Calculation performed assuming 11 operable SRVs; 8 with set pressures of 1131 psig and 3 with set pressures of 1160 psig.

As a result of the SRV set pressure drift in excess of Technical Specification limits, an overpressure analysis (Case 2 and Case 3) has been performed to compare the "as found" performance of the SRVs to the Cycle 3 analysis (Case 1). As in previous operating cycle analyses, Cycle 3 analysis of overpressure protection assumed four SRVs of the lowest pressure group to be out-of-service and the remaining SRVs to lift at +1% of their group's setpoint. The peak reactor steam dome pressure predicted by the Cycle 3 analysis for an overpressure design basis event is 1279 psia. In comparison, the Cycle 2 overpressure analysis predicted a peak pressure of 1267 psia, the difference due primarily to a higher initial reactor vessel pressure in the Cycle 3 analysis. The Technical Specification peak dome pressure limit is 1325 psig (1340 psia).

For the analysis of Cycle 3 "as found" setpoint performance, the SRVs were grouped into three set pressure groups with assigned set pressures which bounded the actual measured setpoint valves. The first group included 8 SRVs with a setpoint of 1131 psig. This setpoint bounds the 1117 psig average set pressure for the lowest 8 SRVs. The second group included 3 SRVs with a setpoint of 1160 psig. This setpoint bounds the 1153 psig average set pressure for the 3 SRVs with recorded set pressures between 1148 and 1159 psig. The third group included 3 SRVs with a set point of 1212 psig. This set point bounds the 1194 psig average set pressure for the 3 SRVs with recorded set pressures between 1178 and 1212 psig.

The results of the analysis show that, even with nine SRVs exceeding the Technical Specification allowable drift of 1%, the resultant peak dome pressure psia was well below the maximum reactor coolant system Technical Specification safety limit of 1340 psia. Further evaluation determined that the Technical Specification vessel pressure safety

limit would not be exceeded during a pressure transient unless eleven SRVs experienced drift on the order of 9%. Accordingly, the consequences to plant safety for the actual setpoint drift experienced was minimal. In the event of a postulated design basis accident, the SRVs would have provided adequate protection for the reactor coolant pressure boundary and assured NSSS integrity.

Corrective Actions:

Previously, Detroit Edison has participated in the BWROG's recommended material substitution strategy by replacing Stellite-6 pilot discs with ferritic stainless steel discs (ARMCO PH13-8Mo). This program was unable to resolve the disc sticking problem and was discontinued by the BWROG in January 1990. Since then, Detroit Edison has completed removal of the PH13-8Mo discs with the last two remaining discs removed during the third refueling outage.

Detroit Edison has also participated in the BWROG SRV Setpoint Drift Fix Committee which initiated a new plan in May of 1990. The BWROG has completed the primary option development under this plan. This option includes replacement of the standard Stellite-6 disc with a Stellite-6 alloyed with 0.3% platinum under a General Electric proprietary process. The nose of the replacement disc (under the seating surface) and the "finger" of the pilot assembly's stabilizer disc assembly are coated with a platinum/palladium catalyst. The function of the coated catalyst and catalyzed disc surface is to reduce the oxygen concentration in the pilot assembly under seat void such that there is less than 0.005 ppm-O_2 in the liquid phase at the pilot disc-seat interface. By test result and theory, the design is capable of reducing the oxygen concentration such that there is virtually no free oxygen present at the metal surface of the disc-seat interface and, therefore, no opportunity for corrosion to take place. To date, there is no in-plant data available to support that the BWROG proposed fix will be successful at solving the SRV setpoint drift problem.

Additional actions taken to minimize setpoint drift include SRV insulation inspections to prevent lower pilot assembly temperature which causes corrosion bonding resulting in setpoint drift. While insulation installation has always been performed in previous outages, the guidelines contained in GE SIL 196, Supplement 16, Revision 1 were

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

proceduralized in the third refueling outage. During this outage, repairs to the Mirror insulation included the addition of pop rivets, buckles and convection seals. Post-work inspections indicated the current insulation configuration and the condition of the insulation will not be a contributing factor to SRV set pressure drift.

The pilot assembly (S/N 341), which did not lift at its mechanical setpoint during as-found testing, has been retested. This pilot was part of SRV B2104-F013E, which is also one of the Automatic Depressurization System (ADS) SRVs during Cycle 3. The results of a test which was conducted to verify that this pilot assembly would function on ADS initiation were inconclusive. The valve will be disassembled for refurbishment and recertification at which time an inspection of the valve internals will be performed to ascertain the most likely failure mode. The results of this inspection are expected to be received and evaluated by January 15, 1993. Based upon the results of the evaluation, if there is a change to the conclusions reached in this Licensee Event Report (LER), a supplement will be submitted.

The referenced SRV pilot assemblies have been replaced by refurbished and certified spare assemblies to reflect the new power uprate setpoint ranges with nominal setpoints of 1135, 1145, and 1155 psig. The removed SRV pilot assemblies will be refurbished and recertified in preparation for installation during the fourth refuelng outage

In addition, Detroit Edison will continue to follow industry developments, including review of the forthcoming Safety Evaluation Report on the BWROG Setpoint Tolerance Relaxation Topical Report (NEDC-31753P).

Previous Similar Events:

Four previous Licensee Event Reports (LER) document similar events:

LER 91-013:	"SRVs Fail Their Set Pressure Test per Technical Specifications"
LER 89-028: (LER 89-028-01)	"Safety Relief Valves Fail Their Set Pressure Tolerance Test"
LER 88-009: (LER 88-009-01)	"Safety Relief Valves Fail Their Set Surveillance Tolerance Test"
LER 86-013:	"Reactor Coolant System Safety Relief Valves Exceed Nameplate Set Pressure Surveillance Test Tolerance"