PRIORITY 1

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ACCESSION NBR:9508290083 DOC.DATE: 95/08/22 NOTARIZED: NO DOCKET #
FACIL:50-335 St. Lucie Plant, Unit 1, Florida Power & Light Co. 05000335
AUTH.NAME AUTHOR AFFILIATION
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SAGAR,D.A. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 95-006-00:on 950810,reactor coolant inventory through shutdown cooling relief valve was lost due to lack of design margin.Increased lift setpoint pressure & reduced minimum required blowdown.W/950822 ltr.

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FPL

Florida Power & Light Company, P.O. Box 128, Fort Pierce, FL 34954-0128

August 22, 1995

L-95-239
10 CFR 50.73

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

Re: St. Lucie Unit 1
Docket No. 50-335
Reportable Event: 95-006
Date of Event: August 10, 1995
Loss of Reactor Coolant Inventory Through a
<u>Shutdown Cooling Relief Valve Due to Lack of Design Margin</u>

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide notification of the subject event.

Very truly yours,

D A Sager

D. A. Sager
Vice President
St. Lucie Plant

DAS/GRM

Attachment

cc: Stewart D. Ebneter, Regional Administrator, USNRC Region II
Senior Resident Inspector, USNRC, St. Lucie Plant

200083

IE22
1/1

NRC FORM 366 (4-95) — U.S. NUCLEAR REGULATORY COMMISSION — APPROVED BY OMB NO. 3150-0104, EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
St. Lucie Unit 1	05000335	1 OF 5

TITLE (4)

Loss of Reactor Coolant Inventory Through a Shutdown Cooling Relief Valve due to Lack of Design Margin

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
08	10	95	95	-- 006 --	00	08	22	95	N/A	
									FACILITY NAME: N/A	DOCKET NUMBER

OPERATING MODE (9)	4
POWER LEVEL (10)	000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)	X	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Kelly J. Korth, Shift Technical Advisor	(407) 465-3550 x3580

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	BP	RV	C710	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)

At 0018 on August 10, 1995, Unit 1 was in Mode 4 in the process of cooling down and depressurizing the Reactor Coolant System (RCS) to investigate the failure of the Power Operated Relief Valves (PORV). The 1A Low Pressure Safety Injection (LPSI) pump was started to initiate flow for Shutdown Cooling (SDC) operation. A thermal relief in the common LPSI discharge piping, lifted during the pump start and did not reseat. SDC operation continued until 0215, August 10, when the lifting relief was discovered and the LPSI pump was stopped.

The root cause of the event was the lack of design margin between the relief valve lifting and reseating setpoints and normal SDC system pressure.

Corrective actions include: 1) The lift setpoint pressure was increased and the minimum required blowdown was reduced, 2) The LPSI thermal relief valve was replaced, and 3) The available design margin for 114 other Safety Related relief valves on both St Lucie units has been evaluated. 17 of these valves will require additional analysis and actions will be taken to increase the margin between system operating pressure and the lift/reseat setpoints where appropriate.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
St Lucie Unit 1	05000335	95	-- 006 --	00	2 OF 5

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

DESCRIPTION OF THE EVENT

On August 9, 1995, Unit 1 was in Mode 4 with both Reactor Coolant System (RCS) (EIIS:AB) loops, and their associated Steam Generators (SGs) available for residual heat removal. The Power Operated Relief Valves (PORVs) (EIIS:AB) had failed to open during stroke testing (Reference LER 335/95-005-00). Per the applicable Technical Specification (TS) action statement, the RCS was to be cooled down and depressurized.

At 0018 on August 10, 1995, with the unit in Mode 4 at 278 degrees and 261 psia, the 1A Low Pressure Safety Injection (LPSI) (EIIS:BP) pump was started to place the Shutdown Cooling (SDC) (EIIS:BP) system in service to continue with the cooldown. Shortly after starting the LPSI pump, utility licensed operators identified that Pressurizer level and Letdown flow were decreasing. The operators did not receive any annunciators normally associated with RCS leakage, did not observe any increase in reactor cavity sump flow and did not detect any level increases in the waste management sumps or tanks (EIIS:WD). Utility non-licensed operators were dispatched to investigate. Inspections of the LPSI pump rooms and other areas in the Reactor Auxiliary Building (RAB) did not identify any leakage. Based on the lack of any confirmatory indications of leakage, the operators concluded that the charging/letdown mismatch was the result of the RCS cooldown. At 0105, the 1B LPSI pump was started and the remaining steps in the SDC normal operating procedure were completed.

At 0215 on August 10, 1995, the control room was notified by the roving fire watch that water was accumulating in the -0.5 ft. elevation of the RAB in the pipe tunnel. Both trains of SDC were immediately secured. The RCS heat removal safety function was being met by the Reactor Coolant Loops and associated Steam Generators. Pressurizer level and charging/letdown flow were observed to be stable, indicating that the leakage had stopped. A control room operator discovered that the RAB (EIIS:NF) floor drain isolation valves to the Safeguards Pump Room sump were closed. When these valves were opened, the high sump level annunciator was received. Visual observation of equipment and piping in the pipe tunnel did not reveal any continuing source of the leakage. Immediately after the event, the flow rate and total amount of leakage was not known. The Emergency Plan Implementing Procedures (EPIPs) were consulted and it was determined that emergency notification was not required.

Based on data evaluated following the event (Charging System makeup water integrator and level increases in the Waste Management System tanks), it was estimated that approximately 4000 gallons of Shutdown Cooling inventory had been diverted to the RAB Waste Management System. The nameplate flow specification of this relief valve is 5 gpm, but communication with the valve vendor, subsequently revealed that the valve had the capability to relieve up to 40 gpm.

At 0611, the 1A LPSI pump was again started and a licensed utility operator, stationed in the pipe tunnel, observed that valve V3439, thermal relief valve in the common LPSI discharge piping, had lifted. The LPSI pump was immediately secured and the relief reseated. At 0940 on August 10, both SDC trains were placed out-of service to replace the thermal relief. Following replacement of the thermal relief, at 0600 on August 11, both SDC trains were restored to operable status. The RCS was then cooled down and depressurized to Mode 5.

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

CAUSE OF THE EVENT

The root cause of the event was inadequate design margin between the relief valve lift and blowdown setpoints and normal SDC operating pressure.

The LPSI common discharge piping thermal relief had a lift setpoint of 500 psig +/- 3% (485 to 515 psig) and a minimum blowdown of 10% (435 to 465 psig). The setpoint bench testing of the relief following its removal, ranged from 480 psig to 500 psig.

The initial operating pressure when establishing SDC is a combination of RCS pressure, LPSI pump differential pressure and a pressure spike due to dynamic forces when the LPSI pump discharge valve is initially opened. Considering that the maximum RCS pressure that the SDC system can be placed in service is 267 psia (SDC suction valves are prevented from opening by an interlock until Pressurizer pressure is below 267 psia), a peak LPSI pump discharge pressure of 487 psig can be developed and a maximum operating pressure of 457 psig can be established.

Therefore, V3439 could lift during SDC initiation since the lift setpoint can be as low as 485 psig and LPSI discharge pressure as high as 487 psig. The valve could then remain open since the reseating point can be as low as 435 psig and the steady state pressure of the SDC system could be as high as 457 psig.

During a unit outage in February 1995 to repair Pressurizer Code Safety Valves on Unit 1, a LPSI suction relief valve lifted when SDC was initiated. A team was assembled to evaluate the event. Based on the results of the evaluation, the SDC initiation procedure was changed. A LPSI pump is started with its discharge valve and all four LPSI injection valves closed. When the discharge valve is open, a downstream pressure spike has been observed. Two injection valves are then throttled opened and a flow of 150 gpm is established and maintained for 15 minutes. The other LPSI pump is started and the remaining valves are throttled open for 5 additional minutes. Flow is then increased slowly. Since the LPSI injection valves are closed when the LPSI discharge valve is opened, this procedure change subjects the LPSI common discharge header thermal relief to a slightly higher dynamic pressure spike than previously experienced. Therefore, this procedure change may have reduced the operating pressure to lift/reseat setpoint margin of this relief valve.

The ability of the operators to detect and mitigate the relief valve lifting was hindered by the Safeguards Pump Room sump isolation valves being closed. On July 31, in preparations for Hurricane Erin, the Safeguards Pump Room sump isolation valves were stroked closed, but not all of the 7 valves controlled by a single switch, had shut. Following troubleshooting efforts, the control switch was allowed to remain in the close position. With the isolation valves closed, the flow path from the LPSI common header thermal relief valve tailpipe to the Safeguards Pump Room sump was isolated. Therefore, the sump high level annunciators were not available to alert operators to the event.

ANALYSIS OF THE EVENT

This event is reportable under 10 CFR 50.73 (a)(2)(vii) as any event where a single cause or condition caused two independent trains or channels to become inoperable in a single system designed to: (a) Shut down the reactor and maintain it in a safe shutdown condition; (b) Remove residual heat; (c) Control the release of radioactive material; or (d) Mitigate the consequences of an accident.

An evaluation was performed to assess the effects of the LPSI common discharge header thermal relief valve lifting on plant operation and safety (Engineering Evaluation JPN-PSL-SENP-95-101). Only SDC operation was considered, where LPSI suction pressure is high enough to challenge the thermal relief. SDC operation with the RCS depressurized or LPSI pump operation during the injection and RCS hot leg recirculation phases of safety injection would have LPSI pump suction pressure sufficiently low such that adequate margin to the relief setpoint would exist.

The capacity of the thermal relief is approximately 40 gpm. Should the relief lift and not reseat during SDC operation, the rate of inventory loss would be well within the charging pump capacity and within the capability of the Waste Management System to remove the water such that Safety Related equipment would not be threatened.

SDC is relied on for long term cooling following certain design basis accidents, specifically: Small Break Loss of Coolant Accidents (SBLOCAs), Excess Steam Demands, and Steam Generator Tube Ruptures (SGTRs). The UFSAR analysis of these design basis accidents involve fuel damage only when considering extremely conservative assumptions. If the conservatism is removed from the analysis, it can be shown that no fuel damage will occur during these events. Therefore, the radiological consequences from these design basis accidents, concurrent with the LPSI common discharge header thermal relief valve lifting, will not be increased and the offsite doses of the UFSAR analysis remain bounding.

Per Technical Specification (TS) 3.4.1.3, with the plant in Mode 4, two of the four heat removal system loops (Reactor Coolant Loops A and B with their associated Steam Generator and at least one associated Reactor Coolant Pump, and SDC loops A and B) shall be operable and at least one reactor coolant or shutdown cooling loop shall be in operation. During this event, the plant was in Mode 4 with both Reactor Cooling Loops operable and the B RCS Loop in operation.

The SDC system is protected from over pressurization during SDC operation by the LPSI suction reliefs. The thermal relief is only required when the system is secured and the portion of piping between the LPSI injection valves and the LPSI discharge check valves is isolated.

Based on the justification listed above, the effect on plant operation due to the lack of design margin between the LPSI common discharge header thermal relief valve setpoint and SDC operating pressures during SDC operation, either during normal plant cooldown or following design basis accidents, was not significant. The health and safety of the public were not affected by this event.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

CORRECTIVE ACTIONS

1) Engineering performed an evaluation to change the LPSI common discharge header thermal relief valve lift setpoint and minimum blowdown to increase the design margin to the systems operating pressure.

2) The LPSI common discharge header thermal relief valve has been replaced with a new relief valve with a lift setpoint of 535 psig and a blowdown range of 6 to 8%.

3) The available design margin for 114 other Safety Related relief valves on both St Lucie units has been evaluated. 17 of these valves require additional analysis and actions are being taken to increase the margin between system operating pressure and the lift/reseat setpoints where appropriate. The results of this review will be made available to the industry via the INPO Nuclear Network.

4) An operator aid is being developed that will provide expected charging/letdown mismatches to maintain a constant Pressurizer level for various cooldown rates.

5) This event will be included into Operations training for both licensed and non-licensed Operations personnel.

6) The Operation Department Supervisor has issued a Night Order reemphasizing the importance of documenting the condition of those components that are in an abnormal configuration in the Valve, Switch Deviation Log.

7) The Emergency Plan Implementation Procedures is under review to determine the appropriate notification threshold for this type of event. Based on this review, changes to the EPIPs will be made if appropriate.

ADDITIONAL INFORMATION

Failed Component Identification

Manufacturer: Crosby Valve & Gage Co.
Model Number: JB 35S-TD SPEC
Device LPSI Common Discharge Header Thermal Relief Valve

Previous Similar Events

LER 335-95-003 described the actuation of a letdown relief that did not reseat until operator action was taken, following an automatic reactor trip.