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SOUTHERN COMPANY
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January 18, 1999

Docket No.: 50-348

NEL-99-0014

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant - Unit 1
Licensee Event Report No. 98-008-00
Reactor Vessel Support Concrete Design Basis
Temperature Exceeded Due To Closed Cooling Damper

Ladies and Gentlemen:

Joseph M. Farley Nuclear Plant - Unit 1 Licensee Event Report No. 98-008-00 is being submitted in accordance with 10 CFR 50.73(a)(2)(ii). There are no NRC commitments in the Licensee Event Report.

If you have any questions, please advise.

Respectfully submitted,

D. Morey

Dave Morey

EWC/maf: ler008.doc

Enclosure

IE22

cc: Southern Nuclear Operating Company
Mr. L. M. Stinson, General Manager

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. J. I. Zimmerman, Licensing Project Manager – Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

NRC FORM 366 (6-1998) U.S.NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	05000348	1 OF 4

TITLE (4)
Reactor Vessel Support Concrete Design Basis Temperature Exceeded Due To Closed Cooling Damper

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
12	23	1998	1998	008	00	01	18	1999	FACILITY NAME	05000

OPERATING MODE (9)	5
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)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2033(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2033(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
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include area code)
L. M. Stinson, General Manager Nuclear Plant	334-899-5156

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	VA	DMP	A340	N					

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-space typewritten lines) (16)

On December 23, 1998, at 2030, with Unit 1 in Mode 5 at 170°F, Unit 1 was found to have been operated outside its design basis. An engineering evaluation determined that the concrete at one reactor vessel support may have exceeded the ASME code temperature limit of 200°F due to loss of forced air cooling. This condition resulted from a reactor cavity cooling system balancing damper being closed. Without forced flow the temperature of the reactor vessel support concrete was calculated to be approximately 280°F. The damper was opened and secured in position, the remaining dampers were verified open and secured in position, and forced flow was verified to be within limits for reactor vessel supports. The evaluation also determined that the concrete could be exposed to temperatures up to 300°F indefinitely without detrimental effects.

The cause of this event was most likely personnel error in that a nut on the actuator arm was either not installed or was improperly installed and subsequently vibrated loose allowing the damper to close. The time of the damper closure and consequently the duration of the misalignment is unknown.

To prevent recurrence, at the next refueling outage on each unit, the reactor cavity cooling system balancing dampers will be verified in the correct positions and measures taken to ensure the dampers remain in the proper position.

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

Plant and System Identification

Westinghouse -- Pressurized Water Reactor
Energy Industry Identification Codes are identified in the text as [XX].

Description of Event

The reactor cavity cooling system consists of fans, dampers, and ductwork and is non-safety-related. A function of the system is to provide forced convection cooling flow through the six reactor vessel supports. Balancing dampers in the ducts ensure proper flow distribution from the individual reactor supports. Cooling of other components in the reactor cavity is provided by the discharge from the containment cooling system. The reactor cavity cooling system is designed to maintain the concrete at the reactor supports within the guidance provided by ASME Code Section III, Division 2 Subsection CC-3440 (a).

On December 23, 1998, at 2030, with Unit 1 in Mode 5 at 170°F, Unit 1 was found to have been operated outside its design basis. A reactor cavity cooling system [VA] balancing damper was found to be closed thus isolating forced air flow to one reactor vessel support. As a result of this condition an engineering evaluation determined that the concrete at one reactor vessel support may have exceeded the FSAR specified temperature of 190°F and the ASME code temperature limit of 200°F. The localized concrete temperature was calculated to be approximately 280°F. ASME Code Section III Division 2 Subsection CC-3440 (a) specifies that localized concrete temperature shall not exceed 200°F. The time of the damper closure and consequently the duration of the misalignment is unknown.

Cause of Event

The cause of this event was most likely personnel error in that a nut on the actuator arm was either not installed or was improperly installed and subsequently vibrated loose allowing the damper to close.

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Safety Assessment

An engineering evaluation was performed based on the guidance provided by ASME Code Section III, Division 2 Subsection CC-3440 (c). Without forced flow the temperature of the reactor vessel support concrete was calculated to be approximately 280°F. The evaluation also determined that the concrete could be exposed to temperatures up to 300°F indefinitely without detrimental effects. It was also determined that the concrete would have performed its support function under normal and accident conditions.

This event would not have been more severe if it had occurred under different operating conditions.

The health and safety of the public was not affected by this event.

Corrective Action

The damper was opened and secured in position, the remaining dampers were verified open and secured in position, and forced flow was verified to be within limits for reactor vessel supports.

To prevent recurrence, at the next refueling outage on each unit, the reactor cavity cooling system balancing dampers will be verified in the correct positions and measures taken to ensure the dampers remain in the proper position.

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

Additional Information

A review of other important Heating, Ventilating, and Air Conditioning (HVAC) systems and operating procedures will be performed to determine whether balance damper position verification is necessary.

The following LERs have been submitted in the last 2 years concerning systems outside the design basis due to personnel error:

LER 98-006-00 Shared, Penetration Filtration Room Filtration System Suction Damper Outside Design and Licensing Basis;

LER 97-013-00 Shared, Operating Outside of Design Basis Due to Control Room Exhaust Isolation Dampers Not Closed;

LER 97-010-01 Shared, Motor Operated Valve Local-Remote Control Circuit Wiring Discrepancies;

LER 97-008-00 Unit 1, Outside of Design Basis Due to RCS Support Gaps Not Being Consistent With Design; and

LER 97-007-00 Shared, Outside of Design Basis Due to Degraded Cork Material.