Dave Morey Vice President Farley Project Southern Nuclear Operating Company P.O. Box 1295 Birmingham, Alabama 35201 Tel 205.992.5131

January 18, 1999

Docket No.: 50-348

SOUTHERN COMPANY
Energy to Serve Your World

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,

By mony

Dave Morey

EWC/maf: ler008.doc

Enclosure

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Page 2 U. S. Nuclear Regulatory Commission

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

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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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U.S.NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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Joseph M. Farley Nuclear Plant - Unit 1	05000348	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	2 OF 4

EXT (If more space is required, use additional copies of NRC Form 386A)(1)

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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U.S.NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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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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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT (If more space is required, use additional copies of NRC Form 386A)(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.