Progress Energy

DEC 15 2009

Serial: HNP-09-130
10 CFR 50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
LICENSEE EVENT REPORT 2009-002-00

Ladies and Gentlemen:

The enclosed Licensee Event Report (LER) 2009-002-00 is submitted in accordance with 10 CFR 50.73, paragraph (a)(2)(i)(B) as a condition prohibited by the plant Technical Specifications, under 10 CFR 50.73(a)(2)(ii)(B) for being in an unanalyzed condition, and under 10 CFR 50.73(a)(2)(v)(B) as a condition that could potentially have prevented the fulfillment of a safety function. This report describes the potential for the Residual Heat Removal Trains to be inoperable during Mode changes. In accordance with 10 CFR 50.73(a) requirements, this LER is submitted within 60 days following discovery of the event.

This document contains no Regulatory Commitments.

Please refer any questions regarding this submittal to Mr. Dave Corlett, Supervisor - Licensing/Regulatory Programs, at (919) 362-3137.

Sincerely,

Kelvin Henderson
Plant General Manager
Harris Nuclear Plant

KH/jmd

Enclosure

cc: Mr. J. D. Austin, NRC Senior Resident Inspector, HNP
Mr. L. A. Reyes, NRC Regional Administrator, Region II
Ms. M. G. Vaaler, NRC Project Manager, HNP

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

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NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

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

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4. TITLE

Potential for Residual Heat Removal Trains to be Inoperable during Mode Change

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	05000
10	16	2009	2009	- 2 -	0	12	15	2009	FACILITY NAME: N/A	DOCKET NUMBER: 05000

9. OPERATING MODE

1

10. POWER LEVEL

100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☑ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☑ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☐ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☑ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
John M. Doorhy Jr. - Licensing Specialist	(919) 362-2137

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
N/A	N/A	N/A	N/A	N					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On October 16, 2009, during a review of Wolf Creek LER 2008-008-002 which addressed potential void formation, a concern with the Harris Nuclear Plant (HNP) Residual Heat Removal (RHR) system during Mode 3 and Mode 4 was identified. During shutdown conditions, HNP typically lines up one or both of the RHR trains to the Reactor Coolant System (RCS) for shutdown cooling and switches them to RHR injection mode of operation prior to entering Mode 3 from Mode 4. During the switchover, system procedures require the RHR system to be cooled down using the mini-flow line through the RHR heat exchanger. The physical location where the mini-flow piping returns to the suction of the RHR system prevents cooling approximately 90 feet of the RHR suction line. The stagnant water in the RHR suction line cools by ambient losses only and can remain high (maximum 350 degrees F) for several hours.

The condition with hot water in the RHR suction piping can also occur shortly after initiating RHR Shutdown cooling mode of operation during the cooldown and would exist until RCS temperatures are reduced sufficiently to preclude a concern. Further review of the plant conditions during the last three years show that the RHR system Limiting Condition for Operation (LCO) has not been met eleven times in Modes 4 through Mode 1.

The root cause is the original RHR system design and system operating guidance provided by Westinghouse were deficient and have allowed conditions to exist where steam flashing can occur in the RHR shutdown cooling piping and pump suction if the RHR system had to be transitioned to the Emergency Core Cooling System (ECCS) mode for injection and recirculation from the shutdown cooling mode in Mode 4 or soon after transitioning to Mode 3.

Immediate corrective actions were to revise plant procedures such that steam void formation in the RHR system would not occur.

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NARRATIVE

Energy Industry Identification System (EIIS) codes are identified in the text within brackets [ ].

I. DESCRIPTION OF EVENT

Industry Operating Experience (OE) addresses conditions discovered at the Wolf Creek plant where Residual Heat Removal system steam voiding vulnerabilities existed at the RHR [BP] pump suction which would affect RHR system operability in Modes 3 & 4 during emergency core cooling system actuation for a Loss of Cooling Accident (LOCA). Review of HNP procedures found that HNP is susceptible to a similar RHR system steam voiding condition. During plant shutdown when transitioning from Mode 3 to Mode 4, both RHR trains could be placed in service for shut down cooling with Reactor Coolant System (RCS) [AB] hot leg temperature as high as 350 degrees F to aid in RCS cool down as well as for chemistry control. When in Mode 4, if an RHR train in service for shutdown cooling had to be switched over to the ECCS [BQ] mode, then the fluid in the RHR pump suction piping could potentially flash to steam if the temperature of the shut down cooling piping is above the saturation temperature resulting from the RHR system pressure when aligned to the Refueling Water Storage Tank (RWST) or to the ECCS sump. If the RHR system was aligned to the ECCS sump, the bounding saturation temperature for the shutdown cooling piping is conservatively estimated to be approximately 183 degrees F. If this were to occur, the resulting steam transient could enter the RHR pump suction, causing the pump to become inoperable. A detailed analysis by an outside vendor will be utilized to determine a more accurate RCS temperature and actions required to prevent flashing in the RHR suction piping.

A similar condition could occur in Mode 4 during plant heat-up because procedures allowed for both trains of RHR to be lined up for RCS cooling. When placing a train of RHR into the ECCS mode in preparation for Mode 3 ascending, procedures require running RHR in the re-circulation mode through the mini-flow line to force cool the system. However, not all of the loop suction piping (approximately 90 ft) from the RCS hot leg can be forced cooled by mini-flow re-circulation to a temperature below the saturation temperature. If a LOCA had occurred requiring ECCS actuation, the hot fluid in the RCS loop shutdown cooling piping on the suction side of the RHR pump could potentially flash to steam resulting in a void at the RHR pump suction which could render the pump inoperable.

The RHR system operating practices have allowed for the potential formation of steam voids in the shutdown cooling piping which could make the system inoperable if the system had to be aligned to the ECCS mode when the plant is in Operational Mode 4 and transitioning to Operational Mode 3. Eleven periods have been identified within the past 3 years when the RHR system LCO was not met and was not in compliance with the requirements of Technical Specification 3.0.4, and Technical Specifications 3/4.5.2 and 3/4.5.3.

In 1993, Westinghouse completed an industry evaluation concerning the potential for water to flash to steam in the RHR pump suction line. Nuclear Safety Advisory Letter (NSAL) 93-004 recommended that plant operating procedures be reviewed to verify that the potential for forming steam voids was precluded. One option presented was to force cool the piping. The Harris plant uses forced cooling, however not all of the loop suction piping (approximately 90 ft) from the RCS hot leg can be forced cooled by the mini-flow re-circulation to a temperature below the saturation temperature of the hot RCS fluid in the line. The stagnant water in the RHR suction line cools by ambient losses only and could remain high (300 to 350 degrees F) for several hours. No records could be found where the NSAL was reviewed by site personnel or if the NSAL recommendations were incorporated into plant procedures which control RCS temperature when placing the RHR system in the shutdown cooling mode in Mode 4 or transitioning to Mode 3. The NSAL also evaluated that the issue did not represent a Substantial Safety Hazard and was not reportable to the Nuclear Regulatory Commission (NRC) per the requirements of 10 CFR 21.

A past operability review by Engineering personnel determined that the total time over the previous three years that RHR would have been inoperable due to this condition using the 183 degrees F value was 217 hours, 59 minutes, with the longest single inoperability being 69 hours, 52 minutes.

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II. CAUSE OF THE EVENT

The root cause of this event is the original RHR system design and system operating guidance provided by Westinghouse were deficient and have allowed conditions to exist where steam flashing can occur in the RHR shutdown cooling piping and pump suction if the RHR system had to be transitioned to the ECCS mode for injection and recirculation from the shutdown cooling mode in Mode 4 or soon after transitioning to Mode 3.

A contributing cause to the event is that HNP did not perform an adequate review of Westinghouse Nuclear Safety and Advisory Letter NSAL-93-004 which addressed the operation of the RHR pumps in the ECCS modes in operational Modes 3 and 4 and provided guidance for preventing the formation of voids in the shutdown cooling piping. The conditions addressed by the NSAL were not fully investigated and therefore the recommendations were not incorporated into plant operating procedures. The inadequate review of NSAL-93-004 resulted in a missed opportunity to implement procedure changes which would have assured that flashing would not have occurred in the portion of RHR shutdown cooling piping that is not forced cooled if RHR had to be aligned to the ECCS mode during Mode 4 or transitioning Mode 3. The NSAL did not address the most limiting condition which is alignment to the ECCS sump. In Westinghouse NSAL-09-008, "Presence of Vapor in Emergency Core Cooling System/Residual Heat Removal System in Modes 3/4 Loss-of-Coolant Accident Conditions", Westinghouse recognized that the saturation pressure used to determine a permissible RHR system operating temperature must consider both the injection and recirculation modes of ECCS operation.

III. SAFETY SIGNIFICANCE

This issue addressed the potential for the development of steam voiding in RHR pump suction shutdown cooling piping if the RHR system had to be transitioned from shutdown cooling to the ECCS mode due to a LOCA occurring in operational Modes 3 or 4. No actual LOCA conditions have occurred so the conditions discussed in this report have never occurred. In Modes 3 or 4, at least one high head Safety Injection pump is available to supply flow to the RCS in a timely manner and the RHR pumps need not be relied on for short-term LOCA mitigation. As a result, a loss of the RHR injection capability during shutdown conditions would pose insignificant consequences. Since the potential exists for steam voids to occur under these conditions and during power ascension, a reportability determination was performed. This reportability determination identified eleven different occasions in the past three years where the LCO requirements for the ECCS subsystems were not satisfied. These violations of Technical Specification requirements are reportable to the NRC under 10 CFR 50.73(a)(2)(i)(B) as operation or condition prohibited by Technical Specifications, under 10 CFR 50.73(a)(2)(ii)(B) for being in an unanalyzed condition, and under 10 CFR 50.73(a)(2)(v)(B) as a condition that could potentially have prevented the fulfillment of a safety function.

IV. CORRECTIVE ACTIONS

Immediate corrective actions to address this issue have been completed to revise Operations procedures to preclude conditions that would lead to void formation in the RHR suction piping.

Planned corrective actions are to utilize the results from a detailed analysis provided by an outside vendor to determine the RCS temperature and actions required to assure prevention of steam flashing in the RHR suction piping when aligning to the ECCS mode including taking suction from the RWST and ECCS containment sump during cool down of the plant, in Mode 4, during heat up of the plant, and finally transitioning to Mode 3. Operations procedures will then be updated to incorporate the results of the detailed analysis.

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V. PREVIOUS SIMILAR EVENTS

No previous similar events have been identified at the Harris Nuclear Plant.