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March 22, 2005 AEP:NRC:2573-25

Docket No. 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
SUPPLEMENTAL LER FOR DEGRADED COMPONENT COOLING WATER FLOW TO CONTAINMENT MAIN STEAM LINE PENETRATIONS

In accordance with the criteria established by 10 CFR 50.73, entitled "Licensee Event Report System," the following report is being submitted:

LER 316/1999-001-01: "Supplemental LER for Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations"

This Licensee Event Report (LER) supplement reports the analysis, root cause, and corrective actions to prevent recurrence of the event.

This supplemental report exceeds the expected submission reporting date estimated in the original LER. This is due to the long-term nature of the work associated with the calculation used to support the conclusions presented in the LER.

There are no new commitments identified in this submittal. Should you have any questions regarding this correspondence, please contact Mr. Toby K. Woods, Compliance Supervisor, at (269) 466-2798.

Sincerely,

Joseph N. Jensen
Site Vice President

HLE/jen

Attachment

IE22

c: J. L. Caldwell – NRC Region III
K. D. Curry – AEP Ft. Wayne
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MDEQ – WHMD/HWRPS
NRC Resident Inspector
Records Center - INPO

NRC Form 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES 6/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 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.

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4. TITLE

Supplemental LER for Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME Cook Plant Unit 1	DOCKET NUMBER 05000-315
06	10	1996	1999	– 001 –	01	03	22	2005	FACILITY NAME	DOCKET NUMBER

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

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Toby Woods, Regulatory Affairs	(269) 466-2798

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	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)

On February 26, 1999, during an engineering review of the Unit 2 Containment System, it was identified that power operation was permitted in June 1996 with degraded Component Cooling Water System (CCW) flow to the coolers for containment penetrations 2-CPN-3 and 2-CPN-4. The main steam lines for steam generators 22 and 23 pass through these penetrations. Operating with the degraded CCW to these coolers may have resulted in excessive thermal stress on the penetration sleeves/liners. This event was reported via a 4-hour Emergency Notification System (ENS) report on February 27, 1999, in accordance with the reporting requirements of 10 CFR 50.72(b)(2)(i), that were in effect at the time the event was discovered, as an event of being in an unanalyzed condition that significantly compromises plant safety. The initial LER was submitted as an event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

This LER supplement reports the final results of the investigation of this event, including root causes, additional analysis, and corrective actions. The causes of this event were deficiencies in the processes for understanding and controlling the design and licensing documents of the penetration coolers and a less than adequate process for initiation, review, and approval of Operability Determinations.

Additional analyses have been performed which determined that the penetrations, including the concrete, sleeves, and liner, were not degraded due to increased local temperatures which resulted from the coolers being out of service. Corrective actions taken include clarification of the UFSAR text that describes the coolers, improvements to the Operability Determination process, a change to the design and licensing basis for the containment concrete to permit local temperatures up to 200 degrees F, and returning all of the main steam penetration coolers to service.

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

Conditions Prior to Event

Unit 2 was in Mode 1 at 100% power.

Unit 1 was in Mode 1 at 100% power.

Description of Event

On February 26, 1999, during the performance of an expanded system readiness review for the Donald C. Cook Unit 2 Containment System [NH], it was identified that a temporary modification had been implemented in August 1996, which allowed for continued power operation without Component Cooling Water System (CCW) [CC] flow to the inner cooling coils [CCL] of the penetration coolers [CLR] for containment penetrations [PEN] 2-CPN-3 and 2-CPN-4. The main steam lines for steam generators [SG] number 22 and 23 pass through penetrations 2-CPN-3 and 2-CPN-4, respectively.

Isolation of CCW to the inner containment cooling coils for penetrations 2-CPN-3 and 2-CPN-4 occurred June 10, 1996, due to failure of valve 2-CCR-441, Containment Penetrations 2-CPN-3 and 2-CPN-4 Inner Cooling Coils CCW Outlet Containment Isolation Valve [ISV]. The valve had failed closed due to a pinhole leak in its diaphragm. This containment isolation valve is normally open, with a failed closed safety function. Prior to maintenance on valve 2-CCR-441, a leak rate test was to be performed which required the coils for 2-CPN-3 and 2-CPN-4 to be drained and pressurized with air at 12 psi. However, the CCW line to the coils was discovered to be blocked. An attempt was made to clear the blockage by pressurizing with water to 135 psi utilizing vent and drain lines, but was unsuccessful.

Blockage of the CCW line between 2-CCR-441 and inner containment cooling coils for penetrations 2-CNP-3 and 2-CNP-4 resulted in loss of cooling flow to those penetration coolers. Valve 2-CCR-441 was repaired; however, due to the blockage, the CCW line to the inner containment cooling coils was declared out of service and the valve left closed. An operability determination (OD) was performed June 14, 1996. To provide additional assurance that the concrete was not being overheated in the local areas around penetrations 2-CNP-3 and 2-CNP-4, concrete temperature data had been collected. Containment exterior concrete surface temperature adjacent to main steam penetrations 2-CNP-3 and 2-CNP-4 was measured, with a high temperature of 155 degrees F. Based upon the temperature data and the assumption that the inner cooling coils were redundant to the outer cooling coils, the operability determination concluded that loss of flow through the penetration cooling coils had no significant impact on the containment concrete temperature in the vicinity of the penetrations.

A safety evaluation was performed August 2, 1996, to assess whether this configuration was acceptable until the next unit outage of sufficient duration to restore the blocked line. This safety evaluation, which relied largely on the June 14, 1996, operability determination, concluded that the CCW supply to main steam line penetrations 2-CNP-3 and 2-CNP-4 satisfied the design cooling requirements in the degraded configuration. The safety evaluation only considered the thermal degradation of the concrete at the penetrations and not the effect of elevated temperatures on the penetration sleeves [SLV] or liners [LNR]. According to the Updated Final Safety Analysis Report (UFSAR), the thermal growth of the penetration sleeves and the stress at the anchors and the liner welds were considered in establishing the penetration temperature limitations. Continued Unit 2 operation without flow to the inner cooling coils for main steam penetrations 2-CPN-3 and 2-CPN-4 created the potential for the containment penetration sleeves and liners to be subjected to excessive heat from the main steam piping. Operating with the degraded CCW to the coolers for these main steam containment penetrations created the potential for excessive thermal stress on the penetration sleeves and liners.

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A similar event occurred at Unit 1 on August 24, 1994. Valve 1-CCR-441, CCW supply to the inner cooling coils for Unit 1 containment penetrations 1-CNP-3 and 1-CNP-4, failed closed due to a diaphragm leak and was declared inoperable. An initial OD was performed August 28, 1994, to assess the impact of lost CCW flow to the penetration coolers. The evaluations performed for failure of 1-CCR-441 at Unit 1 incorrectly concluded that the inner cooling coils on the main steam penetrations were redundant cooling loops to the outer cooling coils. The OD concluded that any combination of two of the four cooling coils would provide sufficient heat removal capacity to keep adjacent concrete temperature within design limits. A follow-up OD was completed on October 28, 1994. This OD concluded that closure of valve 1-CCR-441 would have no adverse effects on the concrete surrounding penetrations 1-CPN-3 and 1-CPN-4. Similar to the initial safety evaluation for the Unit 2 penetrations, the Unit 1 evaluation did not evaluate the impact of degraded cooling on sleeves or liners. The conclusion from the Unit 1 OD performed August 28, 1994, was referenced in the Unit 2 OD of June 14, 1996.

Cause of Event

The causes of the event were:

Deficiencies in the processes for understanding and controlling the design and licensing documents of the penetration coolers. UFSAR, section 5.2.4, previously stated, "Thermal protection of the concrete at hot penetrations is provided by means of redundant cooling coils. Each individual coil is capable of maintaining concrete temperature to a maximum of 150 degrees F. Therefore, in the unlikely event of the failure of one of the coils, the faulty coil can be isolated without loss of thermal protection to the concrete." This wording led the preparers of an OD and 10 CFR 50.59 evaluation to an incorrect conclusion that either the inner cooling coils or the outer cooling coils can provide adequate cooling.

A less than adequate process for initiation, review, and approval of ODs. The conclusions drawn by the preparers of the OD for the Unit 2 CCW cooling coils referenced the previous conclusion that was documented for the Unit 1 CCW cooling coils without verifying initial assumptions to ensure that they were correct.

Analysis of Event

The Emergency Notification System (ENS) report and the subsequent initial Licensee Event Report (LER) report were made in accordance with the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73 that were in effect at the time the event was discovered. This event was initially reported via a 4-hour ENS report at 0012 hours, February 27, 1999, in accordance with 10 CFR 50.72(b)(2)(i) as an event found while the reactor is shut down, that, had it been found when the reactor was in operation, would have resulted in the nuclear power plant, including its principle safety barriers, being in a seriously degraded condition that significantly compromises plant safety. Initial reportability determinations were based on the premise that power operation with degraded cooling to containment penetrations 2-CNP-3 and 2-CNP-4 may have resulted in degradation of the penetration sleeves and/or liners at those locations and rendered them incapable of fully satisfying their functional design requirements under design basis accident conditions. Based on subsequent evaluation of the condition, it was determined that the actual conditions experienced were within the design margins, but outside of the approved design basis of the containment penetrations. Thus, plant safety was not significantly compromised. The initial LER and this supplemental LER are submitted in accordance with the reporting requirements of 10 CFR 50.73(a)(2)(ii), that were in effect at the time of the event, as an event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant.

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Containment piping penetrations are provided for piping passing through the containment walls. The pipe is contained within a sleeve which is welded to the containment liner. In the case of the main steam lines, the pipe is insulated and cooling is provided to limit concrete temperature adjacent to the sleeve. The UFSAR Section 5.2.4 description at the time of the event stated, "Thermal protection of the concrete at hot penetrations is provided by means of redundant cooling coils. Each individual coil is capable of maintaining concrete temperature to a maximum of 150 degrees F. Therefore, in the unlikely event of the failure of one of the coils, the faulty coil can be isolated without loss of thermal protection to the concrete."

The main steam line penetration coolers have two inner cooling coils (inside containment on the flued head) and two outer cooling coils (inside the containment penetration sleeve). Each penetration cooler can perform its design function with one inner coil and one outer coil out of service.

An analysis was performed which determined that, without CCW cooling to the inner penetration coolers for 2-CPN-3 and 2-CPN-4, local concrete temperatures in the vicinity of the main steam line penetrations did not exceed 179 degrees F; however, additional analysis was required to assess whether there was any degradation of the penetration sleeve or liner. Subsequent analysis for this event determined that the resulting stresses in the penetration sleeves, its anchor, welds and the liner were within allowable stresses. Therefore, there was no degradation of the concrete, penetration sleeves, or liner.

The design and licensing basis for the maximum allowable concrete temperature, which is described in UFSAR Section 5.2.4, has been revised in accordance with 10 CFR 50.59 to reflect a maximum allowable concrete temperature of 200 degrees F.

<u>Corrective Actions</u>

1. The CCW flow was restored to all of the main steam penetration cooling coils prior to the restart of each Unit. (Job Orders C0036600 and C0037443)

2. Revised the Operability Determination process and governing procedure to fully incorporate GL 91-18 attributes and add additional rigor to ensure consistent Operability Determination Evaluations. (CRs 98-3176 and 98-6705)

3. UFSAR Section 5.2.4, which describes the design of the containment penetrations, has been revised in accordance with 10 CFR 50.59 to clearly define that each inside and outside main steam penetration cooler has two independent and redundant cooling coils. (CR 99-22293 and UFSAR, Section 5.2.5, revision 19.2)

4. The maximum allowable concrete temperature for normal operating conditions has been increased to 200 degrees F. (CRs P-98-06832, P-00-07070, and 01032027)

<u>Previous Similar Events</u>

None.