Nebraska Public Power District
Nebraska's Energy Leader

NLS2000030
April 4, 2000

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

Gentlemen:

Subject: Licensee Event Report No. 2000-006
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

The subject Licensee Event Report is forwarded as an enclosure to this letter.

Sincerely,

J. A. McDonald
Plant Manager

/lrd
Enclosure

cc: Regional Administrator
USNRC - Region IV

Senior Project Manager
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector
USNRC

NPG Distribution

INPO Records Center

W. Leech
MidAmerican Energy

Cooper Nuclear Station
P.O. Box 98 / Brownville, NE 68321-0098
***Telephone:** (402) 825-3811 / **Fax:** (402) 825-5211*
http://www.nppd.com

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NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If 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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TITLE (4)

Torus to Drywell Vacuum Breaker Misalignment Places Plant in a Condition Prohibited by Technical Specifications

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
03	05	2000	2000	-- 006 --	00	04	04	2000		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	3
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)	X	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)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
S. R. Mahler, Assistant Manager Nuclear Licensing and Safety	(402) 825-3811

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

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)**

On March 5, 2000, at approximately 2200 Central Standard Time, primary containment was declared inoperable due to the failure to meet Technical Specification Surveillance Requirement 3.6.1.1.2, "Verify drywell to suppression chamber bypass leakage is equivalent to a hole less than 1.0 inch in diameter." The acceptance criteria for suppression chamber (torus) to drywell vacuum breaker valve leakage was not met in that the as-found value of 0.51 inches water per minute exceeded the operability value of less than or equal to 0.2 inches water per minute. Investigation revealed that the leakage was caused by improperly orienting the seals in 1997 and pallet/retaining ring misalignment during a previous refueling outage. This was due to inadequate work instructions caused by a failure to identify when the task is beyond the "skill of the craft" and determine when there is a need for additional component-specific guidance or criteria. Corrective Actions include performing necessary repairs using component-specific guidance obtained from the vendor, developing a detailed maintenance procedure, providing training on vacuum breaker maintenance, assessing where additional component-specific guidance/criteria is needed, and evaluating the test procedure to address the impacts different plant conditions may have on test results.

PLANT STATUS

Cooper Nuclear Station (CNS) was in Mode 3, Hot Shutdown, at the time of this event.

BACKGROUND

The torus to drywell vacuum breakers [EIIS:VACB] ensure primary containment [EIIS:NH] integrity by preventing the drywell and suppression chamber (or torus) from exceeding their external pressure design limit.

Under normal operating conditions, the drywell is normally inerted and slightly pressurized above atmospheric pressure, so vacuum relief is not of concern. A primary system rupture, however, results in pressure transients that could challenge primary containment if left unchecked. The function of the vacuum breakers during the early stages of a loss of coolant accident (LOCA) is to initially remain closed to direct steam from the drywell through the downcomers into the torus where it is condensed in the suppression pool. After the steam is condensed, the vacuum breakers may be required to open to ensure the drywell negative design pressure limit is not exceeded.

There are six pairs of internal vacuum breakers (total of 12) located on the vent header of the vent system between the drywell and the torus. The valves are 20 inch vacuum breakers, manufactured by L&J Technologies, Model number LF-240-117. The vacuum breakers are designed to seal and increase in-leakage tightness with increased pressure on the drywell side. Each vacuum breaker is a self actuating valve, similar to a check valve, which can be remotely operated for testing purposes. In order to maintain the valve in the closed position, unless a differential pressure of at least 0.1 psi exists, each valve is equipped with a magnetic latch. The position of each valve is displayed in the control room via a separate position indicating switch. An annunciator will alarm in the control room if any one of the twelve vacuum breakers is not closed.

Maintaining the primary containment pressure suppression function requires limiting the leakage from the drywell to the torus through the vacuum breakers. Leak tests are performed every 18 months to confirm that the equivalent bypass area between the drywell and torus is less than a 1.0 inch diameter hole. This ensures that the leakage paths that would bypass the torus are within allowable limits and the maximum allowable containment pressure would not be exceeded should a LOCA occur.

EVENT DESCRIPTION

On March 5, 2000, at approximately 2200 Central Standard Time, primary containment was declared inoperable due to the failure to meet Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1.2, "Verify drywell to suppression chamber bypass leakage is equivalent to a hole less than 1.0 inch in diameter." The acceptance criteria of surveillance procedure 6.PC.503, "Drywell-to-Suppression Chamber Leakage Test" were not met in that the as-found value of 0.51 inches water per minute exceeded the operability value of less than or equal to 0.2 inches water per minute. TS Limiting Condition of Operation (LCO) 3.6.1.1 Condition A was entered, which requires restoration of Primary Containment to an OPERABLE status within one hour. If this action is not met, the reactor must be placed in Mode 3 within 12 hours and Mode 4 within 36 hours. The reactor was already in Mode 3 at the time this condition was discovered in preparation for a refueling outage. Mode 4 was achieved on March 7, 2000, at 0112.

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

A four-hour non-emergency report was made pursuant to 10CFR50.72(b)(2)(i) and 10CFR50.72(b)(2)(iii)(D).

An inspection of the vacuum breakers revealed five of the twelve vacuum breakers leaking with a variation in severity. Inspections did not reveal any significant degradation of materials nor indications of abnormal wear that would cause the vacuum breakers to leak. Portions of seal material were found to be missing, however it was not in the sealing area and therefore was not determined to be the cause of the leaking vacuum breakers. In addition, the valves were observed during a stroke test to witness pallet impact at closure and to determine if monthly stroke testing could have an impact on valve leakage. It was determined that there was no direct correlation of the pallet slamming closed to the valve leakage.

During Refueling Outage 18 (RE-18) in 1998, corrective and preventive maintenance was performed on the vacuum breakers. In addition, a modification was implemented which changed the bushing material from aluminum to bronze, and the washers from teflon to nylon. Surveillance procedure 6.PC.503, "Drywell-to-Suppression Chamber Leakage Test" was performed at the end of RE-18 with satisfactory results. It should be noted that a differential pressure of approximately 28 inches water was established between the torus and the drywell for performance of this test, whereas a differential pressure of 14 inches water was established for the as-found test performed on March 5, 2000. The different differential pressures used are a function of whether or not primary containment is required and if the atmosphere is inerted at the time the test is performed. This difference in force could affect the sealing ability of the vacuum breakers when they are tested.

Based on inspections performed during the current refueling outage (RE-19), the new materials do not appear to be the cause of the leaking vacuum breakers. There have not been any degraded conditions noted for either the bushing or the washer. It was noted, however that the seals were improperly oriented and the pallets and retaining rings were misaligned. A review of the maintenance and testing history of the vacuum breakers revealed that the incorrect seal orientation likely occurred during corrective maintenance that was performed on the twelve vacuum breakers in 1997. Following the corrective maintenance, as-left leak testing was performed in August 1997 using a differential pressure of approximately 28 inches water. As-found leak testing was performed in September of 1998 at the beginning of RE-18 using a differential pressure of approximately 14 inches water. Both leak tests had satisfactory results. Based on this review and the inspections performed during RE-19, it was determined that the primary source of leakage was the pallet and retaining ring misalignment which occurred during valve maintenance performed in 1998. To ensure the pallet is centered with relation to the body bore, a minimum of 1/16 inch clearance is required between the seal retaining ring and the body bore. This ensures there will not be any physical interference between the two. Two of the vacuum breakers which exhibited the greatest amount of leakage did not meet this criterion. It should be noted that prior to RE-19, the alignment clearance criterion was not known and therefore not a part of the work instructions for maintenance performed during RE-18.

BASIS OF REPORT

The condition identified on March 5, 2000, is being conservatively reported under the requirements of 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications, in that there is evidence that the condition existed before discovery and for a period greater than the TS LCO action completion time.

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

CAUSE

The vacuum breakers failed to meet surveillance test acceptance criteria for the Drywell-to-Suppression Chamber Leakage Test as a result of improperly orienting the seals in 1997 and pallet/retaining ring misalignment during RE-18. This was due to inadequate written instructions caused by the failure of CNS to identify when the task is beyond the "skill of the craft" and determine when there is a need for additional component-specific guidance or criteria. Additionally, the corrective actions from a previous similar event (Licensee Event Report 1997-011), which identified the need for guidance on how to properly perform maintenance on the vacuum breakers, were not fully understood nor properly implemented.

Contributing factors are that the vendor does not have a technical manual regarding maintenance of the valves, it is not easy to align the pallet assembly and working conditions are poor. An additional contributing factor is inconsistencies between the RE-18 as-left and the RE-19 as-found testing.

SAFETY SIGNIFICANCE

There was minimal safety significance associated with this event.

The downcomer pipes in the suppression chamber are equipped with vacuum breaker valves above the suppression pool. These valves are designed to remain shut when drywell pressure exceeds suppression chamber pressure, so that steam from the drywell is directed to the suppression pool and condensed. In the event that the suppression chamber exceeds drywell pressure by approximately 0.5 psi, the valve will open to relieve the pressure differential and prevent backflow of suppression pool water to the drywell. It is essential that the valves remain closed since bypass leakage of steam during a LOCA would reduce the margin between the maximum allowable containment pressure and the calculated pressure response following the LOCA. CNS presented in Amendment 15 of the Final Safety Analysis Report an analysis of allowable bypass leakage over a range of primary coolant system break sizes. CNS determined that the maximum allowable bypass leakage area should not exceed 0.24 square feet (approximately a 7-inch diameter orifice), so that the maximum allowable containment pressure of 62 psig is not to be exceeded. However, for additional conservatism, the bypass leakage limit was reduced to that of a 1.0 inch diameter orifice during surveillance activities. An as-found leakage of 0.51 inches water per minute was identified during the surveillance test and is estimated to give an effective orifice diameter of 2.55 inches. Therefore, although the Technical Specification limit was exceeded, the as-found leakage was less than the maximum allowable leakage that was originally analyzed.

In addition, a Probabilistic Safety Assessment (PSA) review of this event concluded that the extent of the degraded condition had no impact on successful containment system performance (pressure suppression success criteria of the PSA model). The evaluation assumed the condition existed while operating at power. The current model results for Core Damage Frequency and Large Early Release Frequency bound the existing condition. This is based on the fact that the as-found leakage was determined to be well below that which would challenge effective pressure suppression.

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This event was also evaluated to determine if the event should be classified as a Safety System Functional Failure (SSFF). The results of the evaluation demonstrated that CNS retained the ability 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.
D. Mitigate the consequences of an accident.

Therefore, this event is not reportable as a SSFF in accordance with the guidance contained in Nuclear Energy Institute 99-02, Draft Revision D, or under the provisions of 10CFR50.73(a)(2)(v).

CORRECTIVE ACTIONS

Immediate Actions:

Declared primary containment inoperable after failure of Technical Specification Surveillance Requirement 3.6.1.1.2.

Obtained information from the valve vendor, L&J Technologies, to develop work instructions so the vacuum breakers may be maintained with proper guidance. The work instructions were included in a work request already planned for vacuum breaker maintenance. To ensure adequate clearance between the seal retaining ring and the body bore, a 1/16 inch go-no-go gage is to be used.

Performed inspections of the vacuum breakers to identify the causes of valve leakage.

Additional Corrective Actions:

Perform necessary repairs to the torus to drywell vacuum breakers prior to start-up from RE-19.

Develop a detailed maintenance procedure (incorporating the information obtained from the valve vendor) which includes guidance and criteria on how to properly inspect, repair, and align the torus to drywell vacuum breakers.

Provide torus to drywell vacuum breaker maintenance training prior to Refueling Outage 20 (RE-20).

Evaluate 6.PC.503, "Drywell-to-Suppression Chamber Leakage Test," to determine if the different test conditions (drywell pressure when torus is vented, drywell to torus differential pressure, operability limit, etc.) for different plant conditions (primary containment required or not required, etc.) could cause inconsistent leakage results for a similar vacuum breaker physical condition.

Perform a multi-disciplinary assessment to determine where additional component-specific guidance/criteria is needed. If a task is identified that requires a procedure and it currently does not exist, a procedure will be generated to perform the identified task within 120 days following completion of the assessment.

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PREVIOUS EVENTS

Licensee Event Report (LER) 1997-011, "Forced Shutdown Due to Inoperable Torus to Drywell Vacuum Breaker," reported an event where a torus to drywell vacuum breaker could not be confirmed as closed during routine surveillance testing. This resulted in a plant shutdown in accordance with Technical Specifications. The cause was determined to be a failure to maintain the vacuum breakers in accordance with a quality program. The corrective action was to develop a repetitive task under the preventive maintenance program to provide guidance for properly maintaining the vacuum breakers, and to require as-found testing of the vacuum breakers at or near the end of the operating cycle. This task was developed, however subsequent review revealed that it lacked sufficient detail.

Correspondence Number: NLS2000030 (LER 2000-006)

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
Perform necessary repairs to the torus to drywell vacuum breakers prior to start-up from RE-19.	RE-19
Develop a detailed maintenance procedure (incorporating the information obtained from the valve vendor) which includes guidance and criteria on how to properly inspect, repair, and align the torus to drywell vacuum breakers.	August 3, 2000
Provide torus to drywell vacuum breaker maintenance training prior to Refueling Outage 20 (RE-20).	RE-20
Evaluate 6.PC.503, "Drywell-to-Suppression Chamber Leakage Test," to determine if the different test conditions (drywell pressure when torus is vented, drywell to torus differential pressure, operability limit, etc.) for different plant conditions (primary containment required or not required, etc.) could cause inconsistent leakage results for a similar vacuum breaker physical condition.	August 3, 2000
Perform a multi-disciplinary assessment to determine where additional component-specific guidance/criteria is needed. If a task is identified that requires a procedure and it currently does not exist, a procedure will be generated to perform the identified task within 120 days following completion of the assessment.	August 3, 2000 (assessment) December 1, 2000 (procedures)