Nebraska Public Power District
Nebraska's Energy Leader

NLS2001023
March 12, 2001

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

Gentlemen:

Subject: Licensee Event Report No. 2001-001
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

/dnm
Enclosure

cc: Regional Administrator
USNRC - Region IV

Senior Project Manager
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector
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NRC FORM 366 (1-2001) 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)
Cooper Nuclear Station	05000298	1 OF 6

TITLE (4)
Confusing or Incomplete Standards and Administrative Controls Resulted in Failure to Test an Excess Flow Check Valve

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	17	2001	2001 –	001 –	00	03	12	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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

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

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Michael Boyce, Licensing Manager	402-825-5100

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	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 January 17, 2001, it was determined that an excess flow check valve (EFCV) located on the Reactor Vessel head seal leak detection line had not been tested in the last 18 months, as required by Cooper Nuclear Station (CNS) Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.3.8. The EFCV was declared inoperable and the appropriate TS action was entered. This event resulted from decisions based on a past standard (confusing standard) which permitted exceptions described in TS Bases to be used as justification for not complying with a TS SR. Also, the Improved TS conversion project only required the review of new TS Bases which did not ensure that there were no exceptions described in the pre-Amendment 178 TS Bases (incomplete administrative controls).

Immediate corrective actions included completing a review of the individual TS Bases sections effective just prior to Improved TS conversion (Amendment No. 178, issued July 31, 1998) for wording that could lead to a similar TS violation. The long term corrective actions consist of further evaluation of the EFCV to determine its safety function, and to revise the appropriate CNS Policy document(s) to clearly state a TS verbatim compliance policy.

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

PLANT STATUS

Cooper Nuclear Station (CNS) was in Mode 1 at approximately 100 percent power at the time of this event. There were no structures, systems, or components inoperable at the start of the event which contributed to the event.

BACKGROUND

The reactor vessel head seal leak detection system (EIIS: IJ) provides the operator with sufficient indication of reactor vessel top head flange leakage during planned operations. Page 6 of this LER provides a simplified illustration of the system and major components.

A connection on the reactor vessel flange is provided into the annulus between the two metallic seal rings used to seal the reactor vessel and top head flanges. This connection permits detection of leakage from inside of the reactor vessel past the inner seal ring. The connection is piped, via a 1.0 inch carbon steel pipe, to a collection chamber installed between two a-c solenoid-operated valves. The upstream valve is normally open, the downstream valve normally closed. A level switch is provided to detect the accumulation of water in the collection chamber. This level switch actuates an alarm in the main control room. A pressure switch is also provided to actuate the alarm in the main control room as pressure in the leakage collection piping becomes abnormally high. A pressure indicator is provided to indicate the pressure inside the piping arrangement. The level switch is located inside the primary containment, and the pressure instruments are located outside the drywell but inside the reactor building. The instrument pipeline (1.0 inch carbon steel line with a 1/4 inch restricting orifice) for the pressure instruments is provided with one manual isolation valve and one excess flow check valve (EFCV)(NBI-CV-48BCV)(EIIS: JM). The two solenoid valves are controlled by a switch in the main control room. The positions of the valves are indicated by lights. If leakage past the inner seal ring is indicated, the upstream valve can be closed and the downstream valve can be opened by remote-manual operation from the main control room. This action routes the accumulated leakage to the drywell equipment drain sump. After the collection chamber is drained, the solenoid-operated valves can be returned to their normal positions. The leakage rate can be determined by timing the period until the level alarm is reactivated.

The safety function of the excess flow check valve (NBI-CV-48BCV) is to close on excess flow to maintain the reactor coolant pressure boundary in the event of a leak from the reactor vessel flange inner seal (note the outer seal is assumed to be still intact), coincident with a break in the pressure instrument line downstream of the check valve. The open function of the valve (normally open valve remains open) is not a safety function. In the unlikely event that the EFCV is stuck closed, adequate indication of seal leakage remains available to the operator by utilizing the collection chamber referenced above.

The justification for not testing this valve was documented in Technical Specification (TS) Bases Sections 3.7.D and 4.7.D from the initial issuance of the TS in January 1973 until issuance of Amendment No. 178, issued July 31, 1998, which converted the TS and the associated Bases to Improved TS format. These Bases sections stated, in part that "the head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related." (CNS TS Amendment 129, page 184).

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

EVENT DESCRIPTION

On January 17, 2001, while the Check Valve Program engineer was validating that all the EFCVs are tested, it was determined that EFCV NBI-CV-48BCV (EIIS: JM), located on the Reactor Vessel head seal leak detection line (EIIS: IJ), had not been tested in the last 18 months, as required by CNS Surveillance Requirement (SR) 3.6.1.3.8. This SR requires the verification of each reactor instrumentation line EFCV to actuate to the isolation position on an actual or simulated instrument line break every 18 months. This valve cannot be tested due to the system configuration which precludes testing while on-line, Limiting Condition for Operation (LCO) 3.6.1.3, Action C.3 was entered and the affected penetration flow path was isolated. In addition to EFCV NBI-CV-48BCV not being included in the appropriate surveillance procedure for testing, two other EFCVs were identified as also being excluded from the appropriate surveillance procedure. However, these two valves are maintained isolated as locked-closed valves and, therefore, are not reportable in this event.

BASIS FOR REPORT

The event is being reported as a violation of Technical Specifications under 10CFR50.73(a)(2)(I)(B), a condition which was prohibited by the plant's Technical Specifications.

CAUSE

The root cause for NBI-CV-48BCV not being included in the appropriate surveillance procedure and not being tested to comply TS SR 3.6.1.3.8 is that the standards and administrative controls in place were confusing or incomplete. This event resulted from management and technical decisions based on a past standard which permitted exceptions described in TS Bases to be used as justification for not complying with a TS SR (confusing standard). Specifically, the decisions to not test NBI-CV-48BCV were based on the pre-Amendment 178 TS Bases statements going back to the originally issued TS (circa 1973), and that the EFCV was thought not to have a safety function.

In 1973 it was an acceptable practice to use the TS Bases as a place to document clarifying remarks regarding exceptions to TS requirements. Over the years, the Nuclear Regulatory Commission (NRC) has clarified the definition of literal TS compliance. In summary, it is not acceptable to use TS Bases to document an exception to a Technical Specification SR. Recent discussions with personnel in various departments about this condition has revealed that personnel believe that this would not be a condition if ITS would not have not deleted the exception for testing NBI-CV-48BCV out of the TS Bases. Therefore, there is still a misunderstanding about how TS Bases can be used to support TS SRs.

An opportunity to capture this error existed during the Improved Technical Specification (ITS) conversion implemented by Amendment 178. However, the ITS project only required the review of new TS Bases, which did not ensure that there were no exceptions described in the pre-Amendment 178 TS Bases (incomplete administrative controls).

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

SAFETY SIGNIFICANCE

The NRC Safety Evaluation Report cover letter (issued on 14 March 2000) for the Boiling Water Reactor Owners Group (BWROG) General Electric (GE) Topical Report B21-00658-01 "Excess flow check valve testing relaxation" (NEDO-32977-A) cites "a relatively low release frequency estimate in conjunction with extremely low likelihood that this release could impact core damage frequency and negligible consequence of a release in the reactor building."

Omitting the air operated shutoff valve from consideration, it can be stated that a postulated unchecked leak through this line requires the occurrence of a triple compound failure, namely: (1) a leak in the inner vessel head seal, (2) a break of the instrument tubing downstream of the excess flow check valve, and (3) failure of the excess flow check valve to close, all occurring simultaneously.

The likelihood of occurrence for this scenario is very low and the core damage frequency (CDF) is not likely to be affected by any postulated leakage through the line in question. Using an upper bound estimate of f = 1.0E-03 per year for occurrence of an instrument tube break, the corresponding annual probability is P = 1.0 - exp[-1.0E-03] = 1.00E-03. Similarly, using an upper bound estimate of f = 4.0E-02 per year [once in 25 years] for occurrence of a significant inner vessel head seal leak, the corresponding annual probability is P = 1.0 - exp[-4.0E-02] = 3.92E-02. Finally, using an upper bound estimate of f = 1.0E-02 per year for failure of the excess flow check valve to close, a corresponding probability of P = 1.0 - exp[-1.0E-02] = 9.95E-03 results.

Since these three occurrences are independent, the annual probability of occurrence for this scenario is the product of the three constituent probabilities:

P[scenario] = (1.0E-03) (3.92E-02) (9.95E-03) = 3.90E-07.

The value P = 3.90E-07 is below the American Nuclear Standard Institute defined credibility value of 1.0E-06, on an annual basis. This fact, in combination with the fact that conservative, enveloping estimates were used, leads to the conclusion that the significance of the subject scenario is exceedingly low. It should be pointed out that the BWROG GE Topical Report B21-00658-01 "Excess flow check valve testing relaxation" [NEDO-32977-A] shows an upper limit failure frequency of f = 5.53E-03 per year [on page 7] for failure of the excess flow check valve to close. Therefore, the probability of any leakage through the line due, in part, to excess flow check valve failure is negligible.

If leakage did occur, the Control Room operators could isolate the leak by closing an air operated valve. In addition, a manual isolation valve could possibly be closed depending on the location and volume of the leak. If the operators could not isolate the leak, such a postulated leak would be small. This is due to the placement of a 1/4 inch orifice in the instrument line upstream of NBI-CV-48BCV. Any leakage though this orifice would be easily made up with no challenge to the fuel. In addition, any such leakage would not challenge the integrity or functional performance of secondary containment and associated systems. As a result any possible offsite exposures would be well below 10CFR100 limits.

Such a postulated leak would have no impact on the ability of the operators to safely shutdown the plant and it would not place the plant in a condition which would exceed currently analyzed accident conditions or impact any other safety or important to safety functions. Therefore, the safety significance of the reported condition is negligible. The condition did not represent a Safety System Functional Failure, as described in NEI 99-02.

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

CORRECTIVE ACTIONS

Immediate Actions

1. EFCV NBI-CV-48BCV was isolated by closing the isolation valve as required by TS and entering the appropriate LCO (correct condition). Completed January 17, 2001.

2. Performed a review of the individual BASES sections contained in the CNS TS effective just prior to ITS conversion (Amendment No. 178) to determine if there were any other BASES sections that contained language that would appear to establish exceptions to TS requirements. Found three instances where exception wording was not reflected in the associated LCO. Subsequently determined that it was Operations practice to enter the appropriate LCO for all three instances. Therefore, exception wording associated with these three instances had not been used as justification to not enter the appropriate LCO (correct extent of condition and prevent recurrence of root cause). Completed February 21, 2001.

Long Term Actions

1. Determine whether NBI-CV-48BCV has a safety function or not. If determined that it does, determine an alternative method for verifying NBI-CV-48BCV operability and complying with TS SR 3.6.1.3.8. This determination will be made no later than May 31, 2001 (correct condition).

2. Revise the appropriate CNS Policy document(s) to clearly document the TS verbatim compliance policy no later than June 10, 2001 (prevent recurrence of root cause).

PREVIOUS SIMILAR EVENTS

The following is a listing of LERs from the last three years that are a result of the failure to recognize a particular testing requirement, and subsequently not performing the tests or surveillances.

LER 1999-002-00, "Failure to Response Time Test all Reactor Pressure System Results in Missed Surveillance"

LER 1999-005-00, "Failure to Adequately Perform Logic System Functional Testing Places Plant in Condition Prohibited by Technical Specifications"

LER 2000-003-00, "Failure to Meet Logic System Functional Testing (LSFT) Surveillance Requirement When Satisfied by Multiple Surveillance Procedures"

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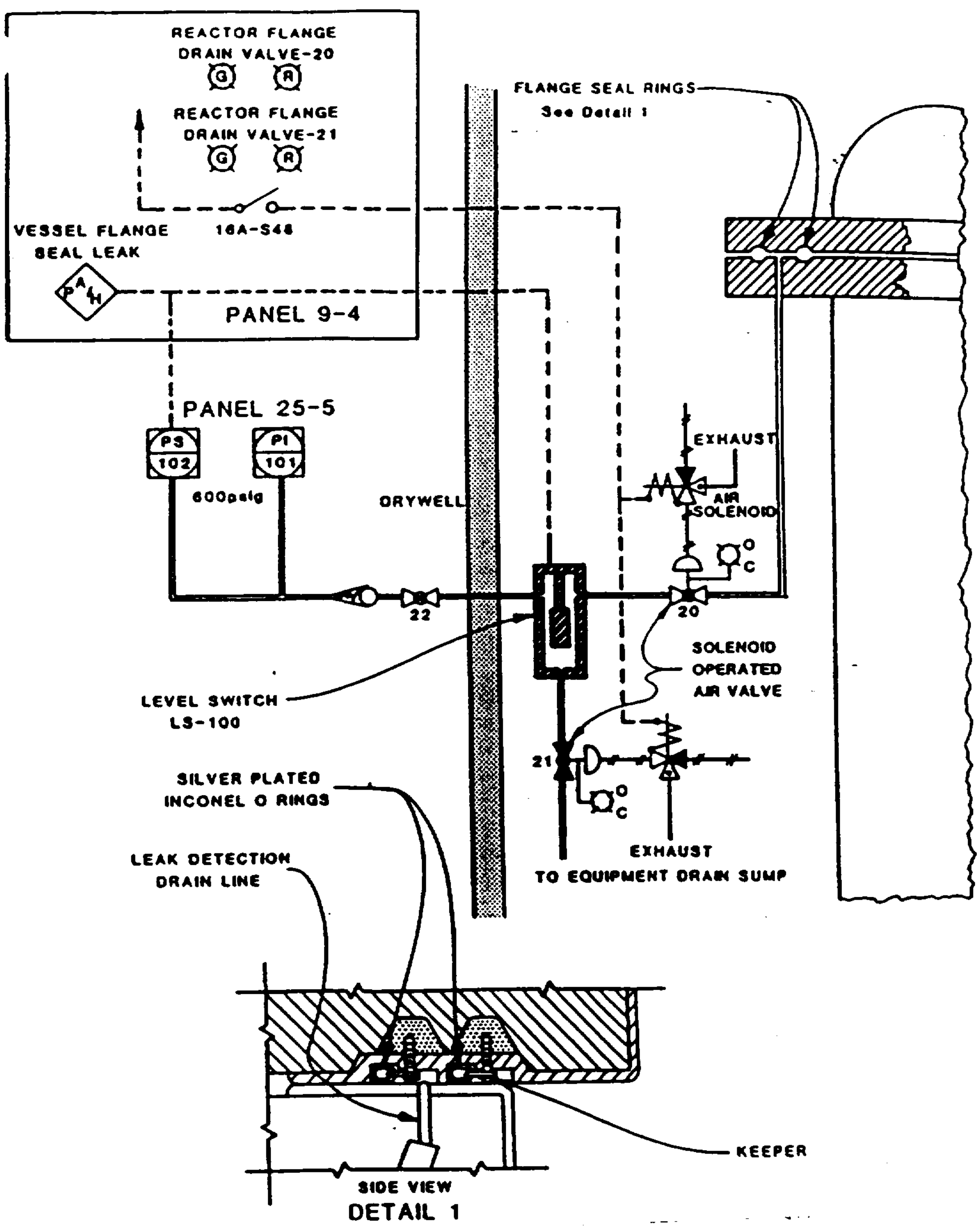
NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

SIMPLIFIED ILLUSTRATION OF REACTOR VESSEL HEAD SEAL LEAK DETECTION SYSTEM



Correspondence Number: NLS2001023

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
Determine whether NBI-CV-48BCV has a safety function or not. If determined that it does, determine an alternate method for verifying NBI-CV-48BCV operability and complying with TS SR 3.6.1.3.8.	May 31, 2001
Revise the appropriate CNS Policy document(s) to clearly document the TS verbatim compliance policy.	June 10, 2001