Nebraska Public Power District

GENERAL OFFICE
P.O. BOX 499, COLUMBUS, NEBRASKA 68601-0499
TELEPHONE (402) 564-8561

NLS8500241

September 3, 1985

Office of Nuclear Reactor Regulation
Operating Reactors Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. D. B. Vassallo, Chief

Dear Mr. Vassallo:

Subject: Containment Purge and Vent Valve Leak Rate Testing

Reference: 1) Letter from D. B. Vassallo to J. M. Pilant dated July 11, 1985, "Request for Additional Information - Purge/Vent Isolation Valve Leak Test Schedule".

2) Letter from J. M. Pilant to D. B. Vassallo dated December 7, 1984, "Containment Purge and Vent System Unresolved Issues".

Reference I requested additional information be submitted to demonstrate that the seals on the purge and vent valves need not be tested on a 3-month schedule. The staff had previously inquired whether the District would be willing to adopt a 3-month or 6-month schedule. The additional information requested is contained in the attachment for your review. Adequate justification is provided for the present 12-month surveillance interval.

In addition to the leak rate testing of the isolation valve seals, other issues regarding the Containment Purge and Vent System remain open. To obtain a final closure of this subject in a timely manner, it is the District's view that all remaining open items should be considered together in a single resolution rather than being treated one at a time. Toward this end, a listing and status of the remaining open items is provided for completeness.

1. Debris Strainers - As noted in Reference 2, the District commits to install debris strainers on the two drywell purge and vent penetrations after satisfactory closure is agreed upon for all remaining issues.

2. Standby Gas Treatment (SBGT) System Use While Purging - The test of an alternate flow path around the SBGT system during inerting on certain start-ups was not conducted this outage due to implementation date contradictions with the Radiological Environmental Technical Specifications (RETS). The District believes the test would have

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verified the feasibility of the alternate flow path. This test will be performed during start-up from the first refueling outage commencing after July 1, 1986. Upon resolution of the other issues, in particular the validity of leak testing the seals on a cycle basis, a proposed technical specification implementing the 90 hour/year restriction will be submitted.

In Reference 2, the District submitted for review an evaluation of the effects on the SBGT system from the gas flow from a Design Basis LOCA through the 2-inch bypass line around the inboard main isolation valve. The evaluation concluded that the SBGT system would remain operable after being subjected to the gas flow, and that, therefore, SBGT system operation through the 2-inch bypass valves need not be constrained by the 90 hour/year limit. The District believes that a review of this evaluation should be a part of the resolution of this issue. Therefore, the proposed technical specification which will be submitted will exclude these 2-inch bypass valves in the 90 hour/year limit.

3. Modification of Three Purge Valves - In the December 15, 1983, submittal, the District submitted proposed modifications to three purge valves to ensure torque values would be in tolerance if the valves had to close against the force of a Design Basis LOCA. The District is awaiting confirmation from the NRC that the proposed modifications are acceptable. When accepted, these modifications will be scheduled to be performed during the following refueling outage.

If you have any questions on the above subject, please call.

Sincerely,

Jay M. Pilant

Jay M. Pilant
Technical Staff Manager
Nuclear Power Group

JMP:grs/km3/6(6A)
Attachment

NRC Request:

1. Describe the type of valve seal testing now being done. Include in your response the testing pressures used and your acceptance criteria.

Response:

Purge and vent valve are tested at 58 psig each operating cycle using air as the test medium. Cooper Nuclear Station operates on a 12-month operating cycle. In no case is the test interval greater than 2 years. The purge and vent valves are tested two at a time with the pressure applied via a test connection which taps into the line between the two valves. The acceptance criteria is 5.0 standard cubic feet per hour (SCFH) for the two valves. A pressure decay method is used to determine the leakage rates in the following steps:

1. Test apparatus is hooked up to the test connection.

2. The volume between the two purge and vent valves, which is a known value, is pressurized to 58 psig with air and then isolated from the test apparatus.

3. The pressure in the volume between the valves is recorded at regular intervals.

4. After determining the pressure decay (psi/min) of the volume, the leakage past the two valves is calculated.

NRC Request:

2. Provide the leak rate data collected to date for the Nordel valve seals. Indicate which leak rate data were collected after the valves were reworked. Also include the age of the seals for each leak rate.

Response:

A detailed leak rate and seal replacement history of the drywell and torus purge and vent isolation valves for the period of 1975-1983 is presented in the following pages. As discussed previously, individual valve leak rates cannot be determined due to the test configuration. The tests give the combined leakage rate for the inboard and outboard valves. It should be noted that four valves (232MV, 238AV, 233MV, and 237AV) have experienced only one failure in the eight years of test data given. Another consideration is that over this period of time the seals were not lubricated to reduce the effects of mechanical wear. A review of old maintenance work requests revealed cases of seal damage due to abrasion and tearing caused by the valve seal. This has contributed to the frequent failure of Valve 245AV since it is cycled every start-up to vent the torus to aid in establishing a drywell-torus differential pressure previously required by Cooper Nuclear Station (CNS) Technical Specifications. This requirement has recently been deleted from the CNS Technical Specifications by Amendment No. 91 to the CNS Facility Operating License. Additionally, in the future CNS plans to lubricate

the valve seals as recommended by the vendor to enhance performance. This should correct the mechanical abrasion phenomena previously observed. The few failures of the valves to pass the leak rate tests has not been attributed to any seal degradation due to environmental conditions during the approximately one-year test interval.

PURGE VALVE SEAL HISTORY

Penetration X-25

Date	232 MV	Local Leak Rate Test Results	238 AV
09/29/75		LLRT-799.7 scfh	
10/01/75			Replaced Seal (MWR 75-9-370)
10/07/75		LLRT- 1.09 scfh	
11/05/76		LLRT- .58 scfh	
12/15/76		LLRT- 1.39 scfh	
04/15/77		LLRT- 1.2 scfh	
05/06/77		LLRT- .17 scfh	
09/20/77		LLRT- 0 scfh	
04/11/79		LLRT- 1.38 scfh	
03/08/80		LLRT- .2 scfh	
05/22/81		LLRT- 1.386 scfh	
05/23/82		LLRT- 3.47 scfh	
05/06/83		LLRT-353.46 scfh	
05/17/83	Replaced Seal (WI 83-1059)		
05/19/83			Replaced Seal (83-1016)
05/20/83		LLRT- 1.73 scfh	

Penetration X-26

Date	231 MV	Local Leak Rate Test Results	246 AV
09/29/75		LLRT-3208 scfh	
09/30/75			Replaced Seal (Seal Damaged) (MWR 75-9-363)
10/14/75	Replaced Seal (Seal Cut) (MWR 75-10-121)		
10/16/75		LLRT- 0 scfh	
10/29/76		LLRT- 2.27 scfh	Replaced Seal (Original seal removed to check seal hardness)
12/16/76		LLRT- 1.89 scfh	
09/21/77		LLRT- α	
09/22/77			Replaced Seal (MWR 77-9-248)
09/23/77		LLRT- 2.52 scfh	
04/03/78		LLRT- .25 scfh	
01/05/79		LLRT- 2.522 scfh	(Tested after installation of new valve actuator)
04/12/79		LLRT- .63 scfh	
05/23/79		LLRT- 2.2 scfh	
05/25/79		LLRT- 1.766 scfh	
03/07/80		LLRT- 60.5 scfh	
	Replaced Seal (MWR 80-3-37)		
03/14/80		LLRT- 3.15 scfh	
05/07/80		LLRT- 3.42 scfh	(Tested after installation of PC-56 MV)
04/28/81		LLRT- 3.78 scfh	
05/25/82		LLRT- 30.65 scfh	
05/26/82		LLRT- 35.19 scfh	
05/27/82		LLRT- 6.05 scfh	
05/06/83		LLRT- 78.7 scfh	
05/10/83		LLRT- 4.83 scfh	

Penetration X-205

Date	233 MV	Local Leak Rate Test Results	237 AV
09/29/75		LLRT- 7.29 scfh	
09/21/77		LLRT- 2.75 scfh	
04/04/78		LLRT-37.18 scfh	
04/04/78	--------------------Tightened Flange Bolts--------------------		
04/04/78		LLRT- 1.93 scfh	
04/10/79		LLRT-80.5 scfh	
04/21/79	Replaced Seal and Gasket	(MWR 79-4-62)	Replaced Seal and Gasket
04/21/79		LLRT- .34 scfh	
03/03/80		LLRT- .9 scfh	
05/26/80		LLRT- .77 scfh	
04/28/81		LLRT- .62 scfh	
05/23/82		LLRT- 1.86 scfh	
05/09/83		LLRT- 1.14 scfh	

Penetration X-220

Date	245 AV	Local Leak Rate Test Results	230 MV
09/29/75		LLRT- 241.92 scfh	
09/30/75	Adjust Seal (MWR 75-9-365)		
10/02/75		LLRT- .75 scfh	
09/20/77		LLRT-1137.96 scfh	
09/23/77			Replaced Seal (MWR 77-9-262)
09/23/77		Trial Leak Rate- 170.39 scfh	
09/23/77	Replaced Seal (MWR 77-9-262)		
09/23/77		LLRT- 3.79 scfh	
04/03/78		LLRT- .76 scfh	
04/11/79		LLRT- 1.4 scfh	
03/03/80		LLRT- 227.6 scfh	
03/06/80	Replaced Seal (MWR 80-3-14)		
03/07/80		LLRT- .91 scfh	
05/09/80		LLRT- 3.8 scfh	(Tested after installation of PC-57 MV)
04/28/81		LLRT- α	
05/02/81	Replaced Seal (WI 81-0839)		
05/04/81		LLRT- 2.8 scfh	
05/24/82		LLRT- α	
06/12/82	Replaced Seal (WI 82-1028)		
06/12/82		LLRT- 2.84 scfh	
10/06/82		LLRT- 2.84 scfh	
05/09/83		LLRT- 113.8 scfh	
05/13/83			Replaced Seal (WI 83-1014)
05/13/83		LLRT- 4.55 scfh	

NRC Request:

3. Indicate the most severe environmental conditions the valves have seen and the most severe environmental conditions the valves may be expected to see. Note: You should indicate both the high and low temperature extremes the valves may see.

Response:

All containment purge and vent valves are located inside the Reactor Building and outside the drywell and torus. An operating temperature of 104°F in the Reactor Building has been established for environmental qualification purposes. Engineering evaluations have revealed no reasons why the non-accident purge valve temperatures should deviate significantly from this 104°F value. The maximum thermal conditions the valves are expected to see for a short period of time during accident conditions are as follows:

Inside Drywell - 340°F
Inside Reactor Building - 285°F