Nebraska Public Power District

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

NLS8400204

July 19, 1984

Office of Nuclear Reactor Regulations
Operating Reactors Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. D. B. Vassallo, Chief

Reference: 1) Letter from J. M. Pilant to D. B. Vassallo dated March 28, 1984, (NLS8400116) "Containment Purge and Vent System Unresolved Issues"

2) Letter from J. M. Pilant to T. A. Ippolito dated December 31, 1981, (LQA8100277) "Containment Purge and Venting System - Proposed Technical Specifications"

Enclosures: a) Evaluation of Purge Isolation Valve Seal Material

b) Example Technical Specification Changes

Dear Mr. Vassallo:

Subject: Containment Purge and Vent System Unresolved Issues
Cooper Nuclear Station (CNS)
NRC Docket No. 50-298, DPR-46

Reference 1 provided the District's plan for resolving the remaining three concerns of the various issues relating to purging and venting the containment. This letter is a follow-up to Reference 1, further describing our resolution of the concerns.

1. Debris Strainers - The District commits to installing debris strainers to insure that isolation valve closure will not be prevented due to debris entrained in the escaping air and steam during a LOCA. These strainers will follow the design previously supplied to the District by the Staff with no additional seismic analysis considered necessary. A schedule for implementation will be provided to our Project Manager at a later date.

2. Leak Rate Testing - Enclosure a is an evaluation of the elastometric material used as sealing surfaces in the purge

A034
1/1

and vent isolation valves at CNS. The District believes this report verifies the resistance to degradation of these seals and justifies a frequency of once per operating cycle for leak rate testing. CNS is on a 12-month operating cycle while Appendix J allows for 24-month tests. To provide additional assurance against seal degradation occurring over a period of several operating cycles, the District will commit to changing out the seals every five operating cycles, even though the manufacturer recommends replacement of the seals every seven years. Example Technical Specification changes associated with this concern are contained in Enclosure b.

3. Standby Gas Treatment (SBGT) System Use While Purging - As stated in Reference 1, this concern is tied in with the Radiological Environmental Technical Specifications (RETS page 216a11; enclosed). The RETS are about to be approved by the Staff, and if certain conditions can be demonstrated, the District can commit to the 90 hour/year limit on SBGT use with reactor coolant temperature >200°F. These conditions involve testing the capability of the alternate flow path around the SBGT system during inerting operations on certain start-ups so that this time does not count against the 90 hours. This is required to minimize the operational constraints imposed on the plant by the 90 hour/year limit. The District currently estimates this verification can be performed by July, 1985. If this test results in verification as expected, the Example Technical Specification Page 165a contained in Enclosure b will be formally submitted to provide closure of this issue.

If you have any questions on this subject, please contact me.

Sincerely,

Jay M. Pilant

Jay M. Pilant
Technical Staff Manager
Nuclear Power Group

JMP/grs:emz19/1
Enclosures

COOPER NUCLEAR STATION
CONTAINMENT PURGE ISOLATION VALVES

VALVE SEAT MATERIAL EVALUATION

1. Purpose. The purpose of this evaluation is to determine the qualification of valve seats in the Drywell and Suppression Chamber purge isolation valves at the Cooper Nuclear Station. In particular, the objective is to determine if the valve seating surfaces containing the elastomeric material EPDM (ethylene-proplyene-diene terpolymer), commercially known as Nordel by DuPont, are capable of resisting degradation over time such that normal valve leakage testing is sufficient to ensure containment integrity.

2. Background. The vaives in question function as containment isolation valves in accordance with 10CFR50, Appendix J, Containment Leakage Testing. As such, Type C (local leakage rate) testing is required once per operating cycle or every 24 months in order to ensure that containment leakage remains within its design limits. In the case of these valves, however, the NRC has requested that valve leakage rates be measured quarterly because the Nordel seating material has not been demonstrated to be sufficiently resistant to degradation to give confidence that leak-tightness will be retained throughout the operating cycle.

3. Valves Being Considered. The valves being considered in this evaluation are:

Vaive No.	Description
PC-MOV-230MV	Suppression Chamber Exhaust Inboard Isolation
PC-MOV-231MV	Drywell Exhaust Inboard Isolation
PC-MOV-232MV	Drywell Inlet Inboard Isolation
PC-MOV-233MV	Suppression Chamber Inlet Inboard Isolation
PC-AOV-237AV	Suppression Chamber Inlet Outboard Isolation
PC-AOV-238AV	Drywell Inlet Outboard Isolation
PC-AOV-245AV	Suppression Chamber Exhaust Outboard Isolation
PC-AOV-246AV	Drywell Exhaust Outboard Isolation

4. Assumptions and General Observations. The following assumptions and general observations are provided:

a. The environment which must be considered in this analysis is the normal environment to which the valves are exposed. The accident environment is not relevant to the basic analysis because any degradation which effects periodic leakage testing frequency results from exposure to normal environmental conditions. For example, periodic valve testing during the operating cycle only provides confidence in valve integrity as of the start of an accident. Periodic leakage testing does not substantiate seal performance when the seals are actually exposed to the more severe accident environment. The above rationale notwithstanding, general conclusions are provided in this analysis regarding the projected performance of Nordel under post-accident conditions.

b. The valves in question are containment isolation valves which are physically located within the Reactor Building. Consequently, the environmental parameters to which the valve seats are exposed are a combination of Drywell environmental parameters and Reactor Building environmental parameters. The parameters used in this analysis are given below:

	Normal	Accident
Temperature	135°F	340°F for 3 hrs 320°F next 6 hrs. 250°F next 24 hrs. 200°F next 6 mos.
Radiation	5.3×10^3 RAD (40 yr. dose)	7.9×10^5 RAD (6 mos. dose)
Pressure	-.5 to 2 psig	70.7 psig (max)
Humidity	90%	100%
Chemical	Nitrogen	Nitrogen

c. The parameters of concern in this analysis are temperature, radiation, and mechanical stress (as a result of seating forces imposed by the valve operators). Pressure, humidity, and chemical effects are not of concern

for the following reasons: (1) Drywell pressure exerted on the butterfly valve balances out its own force since the pressure on half of the valve tends to seat the valve while the forces on the other half tend to unseat it; (2) Humidity does not adversely effect the Nordel as discussed below; and (3) The only chemical the Nordel is exposed to is nitrogen, which slows the aging process as described below.

d. Nordel possesses a high degree of inherent heat resistance. Products made from conventional Nordel can be used in air at temperatures of 250°F to 300°F, with intermittent service to 350°F. In a low oxygen environment (which is the case here due to nitrogen interting), Nordel can be expected to perform at even higher temperatures (References 3 and 4). In addition, steam exposure (and therefore humidity) have negligible effects on Nordel (Reference 3).

e. Tests have shown that the characteristics of EPDM remain substantially unchanged under radiation exposure of less than 10^7 rads (Reference 1). Nordel has been shown to retain 85% of its tensile strength and 55% of its elongation after exposure to 5.5×10^7 rads of beta radiation (Reference 3). These tests have concluded that EPDM is acceptable for usage in nuclear power plants where exposure is limited to 10^7 rads. The BWR Operator's Manual for Materials & Processes (Reference 5) states that the EPDM Nordel by DuPont is acceptable for service up to 300°F and radiation exposure not to exceed 5×10^6 rads. None of the literature reviewed has indicated, based on testing or other analytical evaluation, that EPDM and in particular Nordel, is subject to accelerated degradation at temperatures below 300°F or radiation levels below 5×10^6 rads (References 1 through 9).

f. As an elastomer such as EPDM or Nordel is compressed, there is a change in the elastomer's arrangement and molecular cross-link density. When the compression is released, these changes prevent the seal from returning to its original thickness or size. The loss in dimension, expressed as a percentage of the deflection due to the compression, is called "compression set" (Reference 1).

g. Under continual compression, changes in molecular cross-linking progress to where the seal takes on the shape of the confining surface and the forces exerted by the seal on the confining surface relaxes. By relaxing the compression forces periodically, the seal is allowed to regain much of its original structure so that when it is again compressed, it exerts a renewed force on the confining surface (References 1 and 2).

h. The changes in cross-linking which are reflected in compression set also effect the elongation properties of elastomers such as EPDM. Hence, the results of testing where elongation is measured can be taken as an indicator of compression set (Reference 6). (Note: This assumption pertains to the calculations supporting this analysis.)

i. The valves in question are normally shut valves. Since the Drywell is nitrogen inerted during plant operation, the valves are generally exposed to a low-oxygen, nitrogen-enriched atmosphere.

j. The design of the valves in question is such that the Nordel seating material (on the disc) is compressed by a steel seat, both surfaces of which are essentially flat with gradually sloping transition angles (of approximately 12°).

5. Evaluation. In view of the above discussion, the ability of these valves to remain leak tight is primarily a question of compression set and the capability of the Nordel to retain a sufficient seating force. At the temperatures involved (either the 135°F normal temperature or the sustained temperature at 200°F during the post-accident period) and the radiation levels involved (less than 10^4 rad under normal conditions and less than 10^6 rad under accident conditions), there is no indication that the Nordel is subjected to accelerated degradation This is particularly true in view of the low-oxygen, nitrogen-enriched atmosphere to which the valve seats are exposed. Consequently, this evaluation is reduced to a review of the basic design of the valves, considering the forces applied on the seating areas (by the valve operators) and the data available on compression testing of Nordel.

The maximum seating forces exerted on the Nordel by the valve operating mechanisms are smaller than those forces applied during elastomer testing in accordance with ASTM D395, Rubber Property-Compression Set. The Nordel seat material is stressed by approximately 65 psi under operating conditions while the tests were conducted at stresses nearly twice that amount. Consequently, the total deflection of the Nordel under operating conditions is less than the 25% deflection of the ASTM test (Method B). Under the maximum shaft seating torque, the Nordel valve seal is compressed to approximately 86% of its original thickness. When the valve is reopened, the Nordel returns to approximately 98% of its original thickness. The component of the deflection which is not recoverable (the viscous component) is obviously extremely small and hence the permanent deformation of the elastomer is also quite small. Consequently, since the thickness of the seal is nearly unchanged when the valve reopens, there will be only a minimal effect on the sealing forces during subsequent valve closings. Unless the Nordel is damaged in some way during operation, consistent sealing capability will be provided because essentially constant seating force is retained.

6. Conclusions. The following conclusions are provided:

a. There is no evidence to suggest that satisfactorily leak tested Nordel seats require leak testing more frequently than every operating cycle (required of all containment isolation valves) because of potential time related degradation of the elastomer. Nordel has been shown to sufficiently withstand normal operating temperatures and radiation doses, and will not suffer substantial degradation due to sustained compressive stress from the normally closed valves.

b. There is substantial evidence to indicate that the Nordel will also perform satisfactorily under post-accident conditions. The peak temperature to which the valves may be exposed is above 300°F for only a short period of time and is not considered to be a substantial factor in the performance of the Nordel.

c. Seal performance and expected lifetime can be improved or extended by periodically cycling (opening for several minutes) the valves in order to relax the stress on the seals.

7. References.

(1) Selecting Elastomeric Seals for Nuclear Service, Robert Barbarin, Parker Hannifin Corp/Seal Group (Power Engineering, December 1977).

(2) IE Information Notice No. 84-31: Increased Stroking Time of Bettis Actuators Because of Swollen Ethylene-Propylene Rubber Seals and Seal Set, April 18, 1984.

(3) Radiation Resistance of EPDM, Polychloroprene, and Chlorosulfonated Polyethylene, R.F. Mattia & D.R. Luh, DuPont Elastomer Chemical Dept., (Presented at 1971 Tri-State Regional Meeting of Mass/Conn/RI Rubber Groups, May 13, 1971).

(4) Nordel, Engineering Properties and Applications, DuPont Company, E13193.

(5) BWR Operator's Manual for Materials & Processes, NEDE-20583A, November 1978, Table 2.6, Gasket Materials.

(6) Investigation into Radiation Resistance of Selected Compounds, Robert Barbarin, May 24, 1973.

(7) Materials Science for Engineers, L.H. VanVlack, University of Michigan, 1970.

(8) ASTM D395-82, Rubber Property—Compression Set.

(9) Radiation Effects on Organic Materials in Nuclear Plants, EPRI NP-2129, November 1981.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.C (cont'd.)	4.7.C (cont'd.)

LIMITING CONDITIONS FOR OPERATION

3.7.C (cont'd.)

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. Irradiated fuel is not being handled in the secondary containment.

e. If secondary containment integrity cannot be maintained, restore secondary containment integrity within 4 hours or;

   a. Be in at least Hot Shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

   b. Suspend irradiated fuel handling operations in the secondary containment and all core alterations and activities which could reduce the shutdown margin. The provisions of Specification 1.0.J are not applicable.

D. <u>Primary Containment Isolation Valves</u>

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

SURVEILLANCE REQUIREMENTS

4.7.C (cont'd.)

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$) conditions with a filter train flow rate of not more than 100% of building volume per day. ($\bar{\mu}$= wind speed)

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day, shall be demonstrated at each refueling outage prior to refueling.

d. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

D. <u>Primary Containment Isolation Valves</u>

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times. A valve closure test and verification of closure time shall also be performed prior to returning a valve to service after maintenance, repair or replacement

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7.D (cont'd.)

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolation condition. If the inoperable valve is not returned to the operable status within 72 hours, tag the operable valve closed. Operation may then continue until performance of the next required valve test specified in 4.7.D.1.a.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

4.7.D (cont'd.)

is performed on the valve or its associated actuator control or power circuit.

b. At least once per quarter:

(1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 75%, trip main steam isolation valves individually and verify closure time.

c. At least once per week the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

e. The valve seals of the primary containment and Suppression Chamber Purge and Vent valves shall be replaced at least once per five operating cycles.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.21.C (Cont'd)

7. Containment

a. Whenever the primary containment is vented/purged, it shall be vented/ purged through the Standby Gas Treatment System. With this specification not satisfied, suspend all venting/ purging of the containment. This specification does not apply to Normal Ventilation, or during start-up while performing primary containment inerting in accordance with Specification 3.7.A.5.b following a shutdown of greater than 24 hours.

b. The provisions of Definition J are not applicable. The reporting provisions of Specification 6.5.2 are not applicable.

D. Effluent Dose Liquid/Gaseous

Applicability: At all times.

Specification:

1. The dose or dose commitment to a (actual) member of the public due to radiation and radioactive releases from Cooper Station shall not exceed 75 mrem to his thyroid or 25 mrem to his total body or any other body organ during a calendar year. In the event the calculated dose from radioactive material in liquid or gaseous effluents exceeds two times the limit of Specification 3.21.B.2.a, 3.21.C.2.a, or 3.21.C.3.a, prepare and submit a Special Report, in lieu of any other report, to the Commission pursuant to Specification 6.5.3 within 31 days which 1) defines actions to be taken to reduce releases and prevent recurrence and 2) results of an exposure analysis including effluent pathways and direct radiation to determine whether the dose or dose commitment to a member of the public due to radiation and radioactive releases from Cooper Station during the calendar year through the period covered by the calculation was less than limits stated in this Specification. If the estimated dose exceeds the limits stated herein, and if the condition resulting in doses exceeding these limits has not already been corrected, submission of the Special Report shall be deemed a timely request for a variance in accord with provisions of 40 CFR Part 190, provided

4.21.C.6 (Cont'd)

1) At least once per 31 days during normal operation.

2) Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

b. The radioactivity rate of noble gases at or near the outlet of the main condenser air ejector shall be monitored in accordance with Table 3.21.A.2.

D. Effluent Dose Liquid/Gaseous

1. Dose Calculations - The cumulative dose to a Member of the Public contributed by radioactive material in gaseous and liquid effluents shall be calculated at least once per year in accordance with the ODAM in order to verify compliance with Specification 3.21.D.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.B (cont'd)	4.7.B (cont'd)
4. If these conditions cannot be met, procedures shall be initiated immediately to establish reactor conditions for which the standby gas treatment system is not required.	4.a. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
	b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.
	c. When one circuit of the standby gas treatment system becomes inoperable the other circuit shall be demonstrated to be operable immediately and daily thereafter.
5. Use of the Standby Gas Treatment System for purging/venting the primary containment with both the inboard and outboard exhaust isolation valves open in series is limited to 90 hours per year when coolant temperature is greater than 200°F.	
C. Secondary Containment	C. Secondary Containment
1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.	1. Secondary containment surveillance shall be performed as indicated below:

with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

The intent of Specification 3.7.B.5 is to minimize the time the SBGT system is on line while coolant temperature is greater than 200°F and both inboard and outboard exhaust isolation valves are open in series. The concern is to decrease the probability of damage to the SBGT filters that would occur from excessive differential pressure caused by a LOCA with the main isolation exhaust valves open in series. This specification does allow purge/venting with the bypass around the inboard exhaust valve and the outboard exhaust valve both open in series. The NRC has determined that due to the small size of the bypass valve, there was little chance of damage to the filters if a LOCA occurred while purge/venting the containment with the SBGT system on line.