Nebraska Public Power District

COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
TELEPHONE (402) 825-3811

NLS950029
January 26, 1995

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Cycle Extension
Cooper Nuclear Station
NRC Docket No. 50-298, License No. DPR-46

Reference: Letter (No. NLS940133) to the U.S. NRC Document Control Desk from Nebraska Public Power District dated December 27, 1994; Subject: Exemption Request - 10 CFR 50 Appendix J, Paragraph III.D.2(a)

Gentlemen:

Cooper Nuclear Station (CNS) has been in a forced outage since May 25, 1994. The length of this outage, in combination with the longer than expected 1993 refueling outage and other unplanned outages, has extended the length of the current operating cycle (Cycle 16) for CNS by approximately 12 months. Currently, the Nebraska Public Power District (District) plans to commence the next refueling outage during the month of October 1995.

In support of the extended cycle, the District has reviewed those CNS Technical Specification requirements which require surveillance testing at least once per operating cycle, once every 18 months, refueling cycle, or refueling outage; otherwise categorized as "cycle-related technical specifications." The CNS Technical Specifications define "operating cycle" as the "interval between the end of one refueling outage and the end of the next subsequent refueling outage". Many surveillances are performed on a frequency of at least once per operating cycle. Though not required by CNS Technical Specifications, the District has recently defined "cycle" as 18 months (plus/minus a maximum of 25% interval based on the provisions specified in CNS Technical Specification Definition 1.0.Y) for tracking and scheduling purposes.

In accordance with this internal definition, the District has reviewed both its cycle-related Technical Specification surveillance requirements and non-Technical Specification surveillance requirements for determination as to which of these surveillance requirements should be completed during the current forced outage. Out of approximately 175 surveillance procedures (both Technical Specification and non-Technical Specification) reviewed, the District has elected to perform all but one of the cycle-related Technical Specification surveillances. The District has also elected to perform all but 18 of the cycle-related non-Technical Specification procedures. Written justifications supporting the performing of the 18 surveillance procedures during the 1995 refueling outage have been prepared and are available for NRC inspection.

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The one remaining cycle-related Technical Specification is Surveillance Requirement (SR) 4.6.D.1, which requires, in part, that "approximately half of the Main Steam Safety Valves (SVs) be checked or replaced with bench checked valves once per operating cycle." The specific SV which would, under the internal definition of "cycle", require bench checking during the current outage is MS-RV-70ARV. However, the District has performed a 10 CFR 50.59 evaluation and has determined that adequate justification exists to perform SR 4.6.D.1, as it pertains to this valve, during the 1995 refueling outage. Per discussions with the Nuclear Regulatory Commission (NRC), the District is providing a summary of this justification in the attachment to this letter.

Portions of SR 4.7.A.2.f, "Local Leak Rate Tests" also uses terminology similar with the District's internal definition of the term "cycle" (at least once per operating cycle, once every 18 months, refueling cycle, or refueling outage). However, the schedule requirements for this SR is driven by 10 CFR 50, Appendix J, and requires testing in all cases within two years. For this reason, the District has already requested (see reference) a schedular exemption to implement SR 4.7.A.2.f, as it pertains to the Type B local leak rate test requirements for the Drywell Head and Manport (Penetration X-4), during the 1995 refueling outage. The District will submit a one-time Technical Specification change to support the exemption. Because an exemption request has already been submitted (see reference), it has not been included within the scope of this cycle extension letter.

In keeping with the philosophy that the surveillance interval for cycle-related Technical Specifications is 18 months (+/- 25%), these surveillances will be performed on an 18 month interval unless adequate justification can be otherwise demonstrated. The District has taken a conservative approach in providing your office information regarding its plans to perform SR 4.6.D.1, as it pertains to Main Steam Safety Valve MS-RV-70ARV, during the 1995 refueling outage. The District has determined that no Technical Specification change is required or warranted for this situation.

If you have any questions or need additional information, please contact me.

Sincerely,

John H. Mueller

John H. Mueller
Site Manager

/dnm
Attachment

cc: Regional Administrator
USNRC Region IV

NRC Resident Inspector
Cooper Nuclear Station

NPG Distribution

The following information provides a brief description of the justification supporting the District's decision to perform CNS Technical Specification Surveillance Requirement (SR) 4.6.D.1 during the 1995 refueling outage for Main Steam Safety Valve MS-RV-70ARV. A 10 CFR 50.59 safety evaluation justifying this test schedule and concluding that no unreviewed safety question exists has been approved by the CNS Station Operations Review Committee (SORC).

Component Id: MS-RV-70ARV

Manufacturer: Dresser Type 377QA-RT22

SR Requirement: SR 4.6.D.1 requires, in part, that half of the Main Steam Safety Valves be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

Date last performed: November 5, 1991

District Plan: The District will perform the once per cycle requirement, delineated in SR 4.6.D.1, for Valve MS-RV-70ARV to the 1995 refueling outage, currently scheduled to commence October 1995.

Justification for the District Plan:

Technical Specification 4.6.D.1 requires that approximately half of the SVs shall be checked or replaced with bench checked valves "once per operating cycle". Technical Specification 4.6.D.1 also requires that all SVs be tested "every two cycles". In addition, no work has been performed which would negatively affect Safety Valve performance during the current shutdown. Therefore, no Technical Specification or procedure change is required to support performing this SR during the 1995 refueling outage.

The subject Main Steam Safety Valve (SV) is of Dresser Maxiflow design. The valve components evaluated for this assessment are those subjected to the potentially highly humid Main Steam line internal environment such as internal mechanisms, discs, and pressure boundary components.

The SV utilizes both carbon steel and stainless steel structural materials. Carbon steel is of primary concern for low temperature corrosion influences. However, the critical components of the valves whose function may be affected by corrosion (e.g., discs and seats) are manufactured from corrosion resistant stainless steel. This material is chosen for its overall resistance to corrosion in all potential environments including both cold and operating temperature conditions. These components will not be impacted by the shutdown. For these corrosion resistant components, corrosion virtually stops during the shutdown and will increase upon return to high temperature operating conditions. Therefore, performing maintenance is not necessary as a result of the extended outage.

The length of time a valve may be left installed in a plant prior to its removal is based on Engineering Judgement and experience. The Main Steam Safety Valves are resistant to on-line storage conditions and will be unaffected by the shutdown. They do not have tight clearances subject to the relatively humid steam line environment. The components subject to the inlet environment are made of corrosion resistant materials (i.e., not carbon steel). The SVs do not

require actuation upon return to service and will therefore not be affected by upstream steam line corrosion. Also, the SVs have been tested and will continue to be tested in accordance with ASME Section XI requirements (and within the ASME bounded 5 year frequency requirement).

Surveillance history also supports performing SR 4.6.D.1 for MS-RV-70ARV during the 1995 refueling outage. The most recent surveillance, performed in 1991, test data indicates that lift occurred at 1205 psig which was outside the 1240 psig ± 1% Technical Specification tolerance (NUREG 1433, Standard Technical Specifications utilizes a 3% tolerance). The reduced lift setpoint is an operational concern rather than a safety concern because the drift is in a conservative direction. Substantial margin remain between the normal plant operating pressure of 1000 psig and the as-found setpoint of 1205 psig. A review of the 1989 surveillance data indicated that the valve had minor seat leakage but the as-found setpoint was 1234 psig which was within the ± 1% Technical Specification tolerance.

The District has concluded, as documented in the SORC approved 50.59 analysis, that performing SR 4.6.D.1 during the 1995 refueling outage does not result in an increased probability of an occurrence or increased consequences of an accident previously evaluated in the USAR. The District has also concluded that this plan does not create the possibility of an accident of a different type than that evaluated in the USAR, nor is the margin of safety impacted.

None of the events described in Chapter XIV of the CNS USAR categorized as "accidents" are initiated by the operation, or misoperation, of the SVs. Furthermore, continued use of the SVs (in particular MS-RV-70ARV) following an extended outage; 1) will not degrade the quality of the Reactor Coolant Pressure Boundary component; 2) does not change the required maintenance or inservice inspection intervals; 3) will not create or require any new operator procedures or training; 4) does not add new equipment to a safety-related system or create a new failure mode; 5) does not change an instrument accuracy or response time; 6) does not operate the system outside of its design envelope; 7) does not present a new radioactive leakage path; 8) does not increase a USAR Chapter XIV accident dose calculation result to be greater than its License Acceptance Limit; 9) does not increase onsite or offsite radiation doses; and 10) does not increase personnel hazards. For these reasons, along with additional reasons identified in the SORC approved 50.59 analysis, the District has concluded that performing SR 4.6.D.1 during the 1995 refueling outage will not result in the creation of a safety concern.

Correspondence No: NLS950029

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 Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
Implement SR 4.6.D.1 for Main Steam Safety Valve (MS-RV-70ARV).	During the 1995 refueling outage, currently scheduled to commence October 1995
The District will submit a one-time Technical Specification change to support the schedular exemption for the Drywell Head and Manport.	To support NRC approval of the Technical Specification prior to July 17, 1995 (Current due date of the subject Type B LLRT)