UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 87 TO FACILITY

OPERATING LICENSE NO. DPR-46

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 Introduction

Following the accident at Three Mile Island, Unit 2, the NRC staff developed NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," to provide a comprehensive and integrated plan to improve safety at nuclear power reactors. Specific NUREG-0660 items that were approved by the Commission for implementation were issued in "Clarification of TMI Action Plan Requirements," NUREG-0737. NUREG-0737 and its Supplement 1 (Generic Letter 82-33, "Requirements for Emergency Response Capability") specified that amended Technical Specifications would be required in order to implement several of the items. Subsequently, on January 10, 1983, NRC issued Generic Letter 83-02, "NUREG-0737 Technical Specifications," requesting all boiling water reactor licensees to (a) review their existing Technical Specifications against the GL 83-02 guidance and (b) submit proposed Technical Specifications for those items that deviated from the GL 83-02 guidance.

On April 27, 1983, the Nebraska Public Power District (NPPD) responded to GL 83-02 by submitting to the NRC proposed changes to two sections of the CNS Technical Specifications. The submitted changes are responsive to TMI Action Plan Items II.K.3.3 ("Reporting Safety Valve and Relief Valve Failures and Challenges") and II.K.3.27 ("Common Reference Level").

2.0 Evaluation

A. Reporting SV and RV Failures and Challenges (II.K.3.3)

In their April 27, 1983, letter from J. Pilant to D. Vassallo, NPPD proposed administrative changes to pages 231 and 234 of the CNS Technical Specifications. These changes addressed Action Item II.K.3.3 that was originally discussed in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and further iterated in GL 83-02. According to the submitted changes, NRC will be notified of challenges, malfunctions, or failures of safety and relief valves. All challenges will be reported annually. Malfunctions and failures will be reported by telephone within 24 hours and confirmed by telegraph, mailgram, or facsimile within the first working day after the event with a written followup report within 2 weeks.

We conclude that these commitments are responsive to our request. However, subsequent to the licensee submittal of April 27, 1983, Amendment 86 to the Technical Specifications was issued to incorporate the reporting requirements of 10 CFR Part 50.73. The requirements of 10 CFR Parts 50.72 and 50.73 supersede the requirements of NUREG-0737 relative to reporting safety valve and relief valve failures. Therefore, the reporting requirements on page 234 of the Technical Specifications as proposed by the April 27, 1983 letter have been superseded by Amendment 86 which revised page 231 and deleted page 234. The proposed revision to page 231 to document safety valve and relief valve challenges in the annual reports are unaffected by Amendment 86. We, therefore, conclude that the Technical Specifications proposed by NPPD and updated by Amendment 86, related to Item II.K.3.3, are acceptable.

B. Common Reference Level (II.K.3.27)

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point.

The CNS does not employ a common reference level. However, in a letter from D. Vassallo to J. Pilant dated October 12, 1982, NRC agreed to defer and include the subject item into the control room design review to be performed per NUREG-0737, Action Item I.D.1. The integration of these two action items will lessen the disruption in the control room. In the interim, NPPD has proposed a clarifying revision to the CNS Technical Specification Figure 2.1.1. The revised figure defines the correlation between height above the vessel bottom, instrument reading, and height above the top of active fuel.

We conclude that the revised Figure 2.1.1 is an improvement over the previous figure and inasmuch as the common reference level review is an integral portion of the ongoing review of Item I.D.1 that Item II.K.3.27 is resolved.

3.0 Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and also relates to changes in recordkeeping, reporting or administrative procedures or requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 Conclusions

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Powers

Dated: September 19, 1984

ENCLOSURE 3

STATUS SUMMARY ON NUREG-0737 ITEMS FOR COOPER NUCLEAR STATION

1. STA Training (I.A.1.1.3)

It is an NRC requirement that licensees provide a shift technical advisor (STA) to the shift supervisor to serve the two functions of providing accident assessment and operating experience assessment. As we stated in our letter of April 10, 1980, the Nebraska Public Power District (NPPD) has satisfied this requirement. The training requirements for shift technical advisors are now under consideration by the Commission. Model Technical Specifications that provided NRC staff comments on STA qualifications, training, and on-duty requirements were transmitted to licensees on July 2, 1980; however, no action will be taken to implement NUREG-0737 Technical Specifications for STA training until Commission guidance becomes available.

2. Shift Manning - Overtime Limits (I.A.1.3.1)

On June 15, 1982, the staff issued Generic Letter 82-12, which contained a revised version of the Commission's Policy Statement on nuclear power plant staff working hours. Therein, NPPD was requested to revise the administration section of the Cooper Nuclear Station (CNS) Technical Specifications to adhere to the policy statement guidelines. The objective of the controls are to assure that, to the extent practicable, personnel are not assigned to shift duties while in a fatigued condition that could significantly reduce their mental alertness or their decision making capability. The controls are to be applicable to all plant staff who perform safety-related functions.

Subsequently, NPPD responded on November 24, 1982, by proposing revised Technical Specifications that incorporated administrative procedures for shift overtime. In our letter of April 29, 1983, we found NPPD's submittal to be acceptable, and no further action is required.

3. Dedicated Hydrogen Penetrations (II.E.4.1)

It is a requirement that licensees whose plants use external hydrogen recombiners or purge systems for post-accident control of combustible gas in containments must provide containment isolation systems that are (1) dedicated to ensuring the isolation of those recombiner or purge systems alone, (2) properly sized to permit design flows associated with their respective functions, and (3) redundant and meet the single failure criteria of Criterion 54 and Criterion 56 of the General Design Criteria given in Appendix A to 10 CFR 50.

The CNS plant does not employ hydrogen recombiners but rather utilizes a containment inerting system to prevent the formation of a combustible concentration of hydrogen and oxygen following a design basis accident. As stated in T. Ippolito's letter to J. Pilant dated April 10, 1980, the CNS containment inerting system uses isolation components that are safety related, Seismic Category 1, and fulfill redundancy requirements. Therefore, the CNS containment inerting system satisfies the requirements of Item II.E.4.1.

For post-accident purging of the CNS primary containment, the atmospheric containment atmosphere dilution (ACAD) system is to be used. The ACAD system is designed to be used in conjunction with the standby gas treatment system (SGTS). However, as recognized in Ippolito's letter of April 10, 1980, the ACAD system is not single failure proof when the purge path is through the SGTS because complete reliance would then have to be placed on the integrity of one inboard large containment isolation valve. Subsequently, NPPD reported in J. Pilant's letter to D. Eisenhut dated June 30, 1981, that the CNS purging system has been modified in accordance with the NPPD commitment mentioned in Ippolito's letter of April 10, 1980, and now the venting of combustible gas can be accomplished without opening any large containment isolation valves. Consequently, the issue of dedicated hydrogen penetrations for post-accident combustible gas control of the containment atmosphere was declared resolved by D. Vassallo's letter to J. Pilant dated June 1, 1982, and no further revisions to the CNS Technical Specifications are necessary.

4. <u>Containment Pressure Setpoint</u> (II.E.4.2.5)

Technical Specifications that provide for a minimum containment setpoint pressure which initiates containment isolation for nonessential penetrations and which is compatible with normal operating conditions are currently included in the CNS Technical Specifications. The CNS Mark 1 containment isolation setpoint pressure is 2 psig (drywell to atmosphere). This setting allows about 1 psi for maximum expected containment operating pressure fluctuations; fluctuations that occur due to (1) normal atmospheric barometric variations, (2) momentary pressure surges arising from heat liberated during transient operation of equipment (e.g., pumps) that is located in the drywell, and (3) drift of pressure sensing instrumentation. The 2 psig setpoint also provides about 1 psi to offset instrument error.

The isolation setpoint is high enough that the likelihood of inadvertent containment isolation is minimized. Yet, the limitation on the two major classes of collective variables that constitute the setpoint are more restrictive than permissible values specified in NUREG-0737 (i.e., 1 psi for instrument error) and a subsequent technical evaluation report (i.e., 3 psi for instrument drift and atmospheric variations), which was attached to T. Ippolito's letter to J. Pilant dated October 26, 1981.

We conclude that Item II.E.4.2.5 has been resolved and that no further revisions to the CNS Technical Specifications are necessary.

5. Containment Purge Valves (II.E.4.2.6)

The verification of status, use, testing, and seal maintenance of containment purge and vent valves is an active issue between the licensee and the NRC staff. As stated in J. Pilant's letter to D. Vassallo dated June 23, 1982, NPPD currently abides by the "October 23, 1979 Interim Position for Containment Purge and Vent Valve Operation Pending Resolution of Isolation Valve Operability." Therefore, D. Vassallo's letter to J. Pilant dated November 15, 1982, closed Item II.E.4.2.6 for CNS and no further action is required during this interim period.

Inasmuch as Item II.E.4.2.6 has been incorporated into Multiplant Action (MPA) B-24, "Venting and Purging Containment While at Full Power and Effect of LOCA," the final resolution of this issue will now be completed under the MPA B-24 plant-specific TACS for CNS.

6. Radiation Signal On Purge Valves (II.E.4.2.7)

It is a requirement that containment purge and vent isolation valves close on receipt of a high radiation signal in order to minimize the release of radionuclides to the environs. As prescribed in the CNS Technical Specifications, there are two radiation detectors located in plenums downstream of the purge exhaust valves that will provide such indication. These detectors are set to trip when high radiation is detected (trip setpoint is less than 100 mr/hr).

The use of these detectors and their function was previously approved by D. Vassallo's letter to J. Pilant dated November 15, 1982, and no further revisions to the CNS Technical Specifications are necessary.

7. Reporting SV and RV Failures and Challenges (II.K.3.3)

In their April 27, 1983, letter from J. Pilant to D. Vassallo, NPPD proposed administrative changes to pages 231 and 234 of the CNS Technical Specifications. These changes addressed Action Item II.K.3.3 that was originally discussed in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and further iterated in GL 83-02. According to the submitted changes, NRC will be notified of challenges, malfunctions, or failures of safety and relief valves. All challenges will be reported annually. Malfunctions and failures will be reported by telephone within 24 hours and confirmed by telegraph, mailgram, or facsimile within the first working day after the event with a written followup report within 2 weeks.

We conclude that these commitments are responsive to our request. However, the NUREG-0737 requirements for reporting safety valve and relief valve failures and malfunctions have been superceded by 10 CFR Parts 50.72 and 50.73.

8. RCIC Restart and RCIC Suction (II.K.3.13 and II.K.3.22)

In letters from J. Pilant to D. Vassallo dated April 21, 1983, and August 31, 1982, the licensee described completed modifications to the reactor core isolation cooling (RCIC) system that will (1) restart the RCIC system on subsequent low-water level after it has been terminated by a high-water level signal and (2) automatically transfer the RCIC system suction from the condensate storage tank (CST) to the suppression pool when the CST water level is low. The calibration and surveillance requirements for the RCIC instrumentation are presently addressed in the CNS Technical Specifications, though the RCIC system is not a safety-related system.

In our letters from D. Vassallo to J. Pilant dated June 17, 1983, and May 5, 1983, we approved RCIC system restart and suction switchover modifications, respectively. Subsequently, NPPD proposed in J. Pilant's letter to D. Vassallo of April 27, 1983, that the functional tests for the RCIC system restart and suction switchover be analogous to that of the high pressure coolant injection system and not be explicitly defined in the CNS Technical Specifications. We conclude that the NPPD proposal is acceptable and that Items II.K.3.13 and II.K.3.22 are therefore resolved.

9. Isolation of HPCI and RCIC Modifications (II.K.3.15)

The high-pressure coolant injection (HPCI) and RCIC systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in their systems. Occasionally, pipe break circuitry has been known to result in spurious isolation of HPCI and RCIC systems due to pressure spikes which accompany startup of such systems.

As discussed in the current CNS Technical Specifications, the HPCI and RCIC systems have been modified to employ high-differential pressure actuation timers that briefly delay the respective isolation system instrumentation response, thus precluding inadvertent isolation. Accordingly, Item II.K.3.15 was declared resolved via B. Siegel's letter to J. Pilant dated October 20, 1981, and no further Technical Specification revisions are warranted.

10. Interlock On Recirculation Pump Loops (II.K.3.19)

On plants without jet pumps, interlocks are required to assure that at least two recirculation loops are open for recirculation flow in order that the level measurements in the downcomer region are representative of the level in the core region.

The CNS is a jet pump plant so this item is not applicable.

11. Common Reference Level (II.K.3.27)

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point.

The CNS does not employ a common reference level. However, in a letter from D. Vassallo to J. Pilant dated October 12, 1982, NRC agreed to defer and include the subject item into the control room design review to be performed per NUREG-0737, Action Item I.D.1. The integration of these two action items will lessen the disruption in the control room. In the interim, NPPD has proposed a clarifying revision to CNS Technical Specification Figure 2.1.1. The revised figure defines the correlation between height above the vessel bottom, instrument reading, and height above the top of active fuel.

We conclude that the revised Figure 2.1.1 is an acceptable improvement for reactor water level determination during the interim period until the control room design review is completed. Inasmuch as the common reference level review is an integral portion of the ongoing review of Item I.D.1, then Item II.K.3.27 is closed.

12. Manual Depressurization (II.K.3.45)

Technical resolution of this action item, which deals with the depressurization of the primary coolant system by means other than the automatic depressurization system, is now complete. NRC will not require any modifications in plant design, operation, or Technical Specifications.

Conclusion.

With the exception of NUREG-737 Action Plan Items II.E.4.2.6 and II.K.3.27, which have been incorporated into MPA B-24 and Action Plan Item 1.D.1, respectively, we conclude that NPPD has satisfied all of the Action Plan Items that were delineated in GL 83-02 and that are applicable to the CNS.