NOTICE OF VIOLATION

Nebraska Public Power District
Cooper Nuclear Station

Docket No. 50-298
License No. DPR-46
EA 97-017

During NRC inspections conducted on October 7 through February 19, 1997, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. 10 CFR 50.71(e) requires that the Updated Safety Analysis Report be updated periodically to assure that the information in the Updated Safety Analysis Report contains the latest material developed. 10 CFR 50.71(e)(4) requires that revisions be filed annually or 6 months after each refueling outage provided the interval between successive updates does not exceed 24 months. The revisions must reflect all changes up to a maximum of 6 months prior to the date of filing.

   Contrary to the above, the licensee did not update the USAR within the required timeframe for the following examples, each of which constitutes a separate violation.

   1. As of November 1, 1996, Updated Final Safety Analysis Report Section XII-2.3.5.2.2, "[Seismic Analysis] Piping," and Updated Safety Analysis Report, Appendix C, "Structural Loading Criteria," Section 3.3.3.2, "Piping Seismic Analysis," was not updated to accurately reflect the seismic analysis practices at the time. Since initial construction of the facility, these sections of the Updated Safety Analysis Report (and the Final Safety Analysis Report), have described, in detail, the procedure for dynamically analyzing Class-I seismic piping systems without restricting the requirement for dynamic analysis to large bore piping. However, as of November 1, 1996 (and since initial construction), the dynamic seismic analysis described in the Updated Safety Analysis Report was not performed for 2-inch and smaller piping systems.

   2. As of November 1, 1996, Updated Safety Analysis Report, Section III-9.3, "[Standby Liquid Control System] Description," was not updated to accurately reflect the expected room temperatures for the standby liquid control system and the controls in place to ensure safe operation with a room temperature of 50°F. Updated Safety Analysis Report, Section X-10.3.2 "[Heating, Ventilation and Air Conditioning Systems] Station Heating System," states that winter design temperatures for the system are given in Table X-10-1. Table X-10-1, "[Heating, Ventilation and Air Conditioning Systems] Station Heating System Design Temperatures (Winter)," states that the normal minimum indoor temperature for the reactor building is 50°F. The equipment containing the solution is installed in a room in the reactor building. However, as of November 1, 1996 (and since initial construction), Updated Safety Analysis Report, Section III-9.3,

"[Standby Liquid Control System] Description," stated that, "The equipment containing the solution is installed in a room in which the air temperature is to be maintained within the range of 65°F to 100°F."

3. As of November 1, 1996, Updated Safety Analysis Report, Section III-9.4, "[Standby Liquid Control System] Safety Evaluation," was not updated to correctly describe the safety basis for the standby liquid control system relief valves. Design Change 86-34A, "SLC/ATWS Modifications," Revision 0, dated March 4, 1988, changed the safety basis for the standby liquid control system relief valve settings from:

    assuring injection into the reactor above the ***normal*** [emphasis added] pressure of approximately 1030 psig in the bottom of the reactor,

to:

    assuring injection into the reactor above the anticipated transient without scram reactor pressure conditions, which would equal the reactor safety/relief valves' setpoints plus the accumulation at the maximum anticipated transient without scram steam flow, (i.e., approximately 1100 psig plus the static head in the reactor vessel).

Specifically, Section III-9.4 continued to state that "The SLC system and pumps have sufficient pressure margin, up to the allowed system relief valve setting range of 1450 to 1680 psig, to assure solution injection into the reactor above the ***normal*** [emphasis added] pressure of approximately 1030 psig in the bottom of the reactor."

4. As of November 1, 1996, Updated Safety Analysis Report, Section III-9.3, "[Standby Liquid Control] Description," was not updated to be consistent with Technical Specification Figure 3.4.2. The Updated Safety Analysis Report states that, at the minimum room temperature of 65°F, the maximum permitted solution concentration is 12.5 weight percent. Section III-9.3 also states that a concentration of 11.5 percent corresponds to an adjusted saturation temperature of 61°F. The Updated Safety Analysis Report adjusted saturation temperature includes a 10°F margin over saturation, which corresponds to the definition for the Technical Specification minimum allowable temperature. However, Technical Specification Figure 3.4.2, "Percent Sodium Pentaborate by Weight of Solution versus Temperature," indicates that at 65°F, the maximum permitted concentration was 12.1 percent. At 11.5 percent concentration, the minimum allowable temperature was 62°F.

5. As of November 1, 1996, Updated Safety Analysis Report Section IV-9.3, "[Reactor Water Cleanup System] Description," was not updated to clearly indicate the effect of a modification on the reactor water cleanup system isolation valves' control logic. Further, neither Section IV-9.3 nor

Section III-9.3, "[Standby Liquid Control System] Description," were updated to indicate that following the modification it was always necessary to operate both SLC trains to close both motor operated valves and maintain comparable defense against a single failure of the reactor water cleanup isolation valves.

At the time of the inspection, Updated Safety Analysis Report, Section IV-9.3, "[Reactor Water Cleanup System] Description," stated that, "In the inlet piping to the cleanup recirculation pumps, two motor operated isolation valves, one on either side of the primary containment, are automatically closed by... standby liquid control system actuation." Design Change 86-34A, "SLC/ATWS Modifications," Revision 0, March 4, 1988, changed the reactor water cleanup system isolation valves' control logic such that initiation of one train of the SLC system no longer closed both motor operated valves. In order to achieve comparable defense against a single failure of the reactor water cleanup isolation valve, the licensee implemented administrative procedures which require the operators to always operate both trains of the standby liquid control system.

6. In 1994, during the surveillance test validation program status review, the licensee identified at least two discrepancies in the Updated Safety Analysis Report, which were not corrected in the July 22, 1996 update to the Updated Safety Analysis Report.

   (a) Updated Safety Analysis Report, Table VII-3-1, "Pipeline Penetrating Containment," Note 4 incorrectly stated that the control rod drive system solenoid valves open during a reactor scram. On reactor SCRAM the solenoid valves remain closed and the air-operated SCRAM valves open to insert the control rods and to exhaust water to the SCRAM discharge volume.

   (b) Updated Safety Analysis Report, Section VII-4.5.44, "[Core Spray System Control and Instrumentation] Core Spray Valve Control," incorrectly stated that two pressure switches monitor system pressure (for the low pressure permissive). In addition, it indicates that either switch can initiate opening of the discharge valves for core spray. There actually are four pressure switches designed in a 1-out-of-2 twice logic and a minimum of two switches are required to actuate to initiate opening of the core spray valves.

7. As of March 16, 1996, Updated Safety Analysis Report, Table V-2-2, "Penetration Schedule," pages V-2-9 to V-2-12, was not updated to correctly list all the penetrations; the quantity of lines in three penetrations; and line descriptions in five penetrations.

8. As of May 4, 1996, Updated Safety Analysis Report, Table V-2-7, "Testable Primary Containment Isolation Valves," pages V-2-44 to V-2-46, did not list 23 penetrations (X20, X-30E and -30F, X-33E and -33F, X-35A through E, X-45D, and X-229A through L) and their associated valves.

B. 10 CFR 50.59(b)(1) requires that the licensee maintain records of changes in the facility and of changes in procedures made pursuant to this section, to the extent that these changes constitute changes in the facility as described in the safety analysis report or to the extent that they constitute changes in the procedures as described in the safety analysis report. Further, these records must include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question.

Contrary to the above, the licensee either did not perform the required written safety evaluation or performed an inadequate safety evaluation as shown in the following examples, each of which constitutes a separate violation.

1. Updated Safety Analysis Report, Section X.8.2.8.C, "Common Mode Failure Analysis - Fire," required that combustibles not be located in the service water booster pump room area since both trains of the service water system were located in close proximity. However, Procedure 0.7.1, "Control of Combustibles," Revision 6, allowed up to 90 pounds of wood or 5 gallons of flammable liquid for this area of the plant. In addition, on December 2, 1996, combustible materials (rags, papers, and flammable chemicals) were located in the service water system booster pump room. The written safety evaluation for this change in procedure was inadequate, in that, a common mode failure analysis had not been performed to justify the presence of combustible materials in the service water booster pump area, and therefore, the safety evaluation did not provide a bases for the determination that the change does not involve an unreviewed safety question.

2. Updated Safety Analysis Report, Section XII.2.2.7.1, "Intake Structure," states, in part, that in order to keep ice away from the intake structure during cold weather, an ice deflector is installed during the winter months. Although a portion of the ice deflector was installed on December 18, 1996, the deflector had not been fully installed at any time during the winter months. The failure to fully install the ice deflector by the winter months was a configuration changes that had not been evaluated, through a written safety evaluation, as a change to the facility.

3. Updated Safety Analysis Report, Section IV.10.3, "Nuclear System Leakage Rate Limits - Description," states, in part, that each containment drywell sump has an alarm system and automatic starting sequence on rising water level. Both containment drywell sumps are equipped with a fill rate timer and alarm. This alarm can be set at or below the Technical Specification limits and would provide immediate indication when this preselected rate is reached or exceeded. However, the safety evaluation dated December 20,

1996, that addressed the failure of the automatic pump starting system, and the failure of the sump fill rate timer and high level alarm, was inadequate in that it did not address the lack of control room alarm. A separate written safety evaluation did not exist for this change to the facility.

These violations represent a Severity Level III problem (Supplement I) (50-298/96024-14).

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Arlington, TX
this 25th day of June 1997

ENCLOSURE 2

On April 15, 1997, the management of the Nebraska Public Power District and the NRC met in a predecisional enforcement conference to discuss the apparent violations which were identified in NRC Inspection Reports 50-298/96-24 and 50-298/96-31. The licensee did not agree that four of the examples presented at that conference were violations of 10 CFR 50.71(e). They also did not agree that one of the examples presented was a violation of 10 CFR 50.59(b)(1). The contested examples, the licensee's view, and the basis for the NRC's conclusion are described below.

10 CFR 50.71(e)

10 CFR 50.71(e) requires that the Updated Safety Analysis Report (USAR) be updated periodically to assure that the information in the USAR contains the latest material developed. Contrary to the above, the NRC identified the following examples of apparent violations of 10 CFR 50. 71(e):

Contested Example 1

USAR, Section VII-2.3.5.1, "Alternate Rod Insertion (ARI)," indicated that the time delay for initiating alternate rod insertion should be applied only to the low reactor water level initiation signal and not for the high reactor pressure initiation signal. This was consistent with Drawing 945E632, "ARI/ATWS Recirc Pump Trip," Sheets 3 and 7, dated January 15, 1995. However, as of November 1, 1996, USAR, Section III-5.5.3.4, "Alternate Rod Insertion," was incorrectly updated to state that the time delay for initiating alternate rod insertion should be applied to both the low reactor water level initiating signal and the high reactor pressure initiating signal.

Licensee's View

During the predecisional enforcement conference, the licensee stated that they were not in violation of 10 CFR 50.71(e) at the time of NRC Inspection 50-298/96-24. Cooper personnel acknowledged that in 1993, the USAR had been inaccurately updated following the modification to install alternate rod insertion capability. However, licensee personnel had detected the error and initiated a change to the USAR to correct the inaccuracy prior to the arrival of the NRC inspection team.

Basis for NRC Conclusion

The NRC agreed that the licensee was not in violation of 10 CFR 50.71(e) at the time of the inspection. Since the licensee had already corrected the violation which occurred in 1993, this example was not included in the final citation.

Contested Example 2

Design Change 86-34A, "SLC/ATWS Modifications," Revision 0, dated March 4, 1988, indicated that the standby liquid control system (SLC) relief valve settings were intended to assure injection into the reactor above the anticipated transient without scram reactor pressure conditions, which would equal the reactor safety/relief valves' setpoints plus the accumulation at the maximum anticipated transient without scram steam flow, (i.e., approximately 1100 psig plus the static head in the reactor vessel). However, as of November 1, 1996, USAR, Section III-9.4, "[Standby Liquid Control System] Safety Evaluation," had not been accurately updated. Specifically, Section III-9.4 continued to state that, "The SLC system and pumps have sufficient pressure margin, up to the allowed system relief valve setting range of 1450 to 1680 psig, to assure solution injection into the reactor above the *normal* [emphasis added] pressure of approximately 1030 psig in the bottom of the reactor." The original standby liquid control system design was based on normal operation.

Licensee's View

The licensee offered two reasons for the acceptability of the current USAR wording.

1. The licensee believed that the stated information accurately described the design basis for the standby liquid control system. They noted that anticipated transient without scram (ATWS) was specified as a beyond design basis event in several licensing documents. In their view, the license basis was separate from the design basis and it was appropriate to preserve information about both in the USAR.

2. NPPD believed that to update this paragraph to also describe the license basis (i.e., need to comply with the 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants") was beyond the level of detail which was required for the USAR.

Basis for NRC Conclusion

The NRC disagreed with the licensee's positions and included this example in the final citation.

1. The NRC acknowledged that ATWS was not a Chapter 14 design basis event for the Cooper Nuclear Station. As such, the licensee was not required to implement all of the design requirements which would normally apply to equipment needed to mitigate a design basis event (for example full 10 CFR Part 50, Appendix B, requirements or full single failure analysis). However, the NRC determined that the ability to mitigate an ATWS was part of the design basis as defined in 10 CFR 50.2. In the 10 CFR 50.2 sense, the design basis is that information which identifies specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chose for controlling parameters referenced in the design. During rule making, 10 CFR 50.62 back fit onto all licensee's requirements to ensure facilities would be capable of mitigating an ATWS. The bounding changes in parameters which are necessary to combat an ATWS constitute the new design basis of the facility and the USAR should clearly describe the bounding parameters.

   The NRC noted that Section III-9.4 was the safety evaluation for the standby liquid control system. The NRC determined that the licensee's policy of trying to separately track the "license basis" and the "design basis" provided nonconservative and confusing information regarding safety basis for the facility. It was not evident from reading this portion of Section III-9.4 that the standby liquid control system and pumps are required to have sufficient pressure margin, up to the allowed system relief valve setting range of 1450 to 1680, to assure solution injection into the reactor above the reactor safety/relief valves' setpoints plus the accumulation at the maximum anticipated transient without scram steam flow. The nonconservatism could easily lead to an incorrect operability determination or an incorrect 10 CFR 50.59 evaluation.

2. The NRC also disagreed with the licensee's level of detail argument. The NRC determined that the level of detail was already established with the existing paragraph. The NRC expects that the facility be accurately discussed (i.e., the bounding safety parameters be described) at the current level of detail.

Contested Example 3

Technical Specification Figure 3.4.1 and Calculation NEDC 93-142, Revision 0, dated August 2, 1993, "SLC Storage Tank Setpoints and Concentration Requirements," required the minimum volume of sodium pentaborate solution in the standby liquid control system storage tank to be between 3132 and 4414 gallons for concentrations between 16 and 11.5 percent respectively. USAR, Section III-9.4, "[Standby Liquid Control System] Safety Evaluation," and Technical Specifications 4.4.A.1 and 4.4.A.2.b, "Standby Liquid Control System," stated that the minimum required flow rate for the standby liquid control system pumps was 38.2 gpm. Based on these values, the minimum time for one pump to inject the required volume of solution would have been 82 to 116 minutes, and 41 to 58 minutes for two pumps. However, the following USAR sections were not correctly updated when the standby liquid control injection criteria was changed to meet the ATWS rule:

1. USAR, Section III-9.3, "[Standby Liquid Control System] Description," stated that, "Each positive displacement pump is capable of injecting the required weight of solution into the reactor in 53 to 120 minutes, independent of the amount of solution in the tank (within the required volume), and the pump rate (within the specified Technical Specification limits)."

2. USAR, Section III-9.4, "[Standby Liquid Control System] Safety Evaluation - Compliance with the NRC ATWS Rule," stated, "With the simultaneous operation of both positive displacement pumps, the solution can be injected into the reactor in 27 to 60 minutes, independent of the amount of solution in the tank (within the required volume), and the pump rates (within the specified Technical Specification limits)."

Licensee's View

The licensee stated that they agreed that this section accurately reflected the system design basis. CNS has always understood that this text to be the design basis requirements. They determined that it was an issue of clarity and noted that the NRC had interpreted the text to be system capability not design basis requirements. They stated that they had clarified the section but that they did not believe it was a violation of 10 CFR 50. 71(e).

Basis for NRC Conclusion

The NRC agreed with the licensee's presentations and this example was not included in the final citation. The need for clarification was discussed in NRC Inspection Report 50-298/96-24.

Contested Example 4

Alarm Procedure 2.3.2.28, "Panel 9-5-2, Window F-8, [Standby Liquid Control] SLC Tank Hi/Low Level," Revision 26, dated March 15, 1996, specified the high and low alarm setpoints for the standby liquid control system sodium pentaborate storage tank as 84 and 74 percent respectively. Surveillance Procedure 6.SLC.601, "[Standby Liquid Control] SLC Tank Sampling," Revision 0, dated November 17, 1995, indicated that these setpoints corresponded to 3835 and 3378 gallons, respectively. However, USAR, Section III-9.3, "[Standby Liquid Control System] Description," inaccurately stated that these alarms were set at 3850 and 3350 gallons, respectively.

Licensee's View

The licensee stated that the USAR should be clarified and that the values presented were nominal values. The actual settings were determined using the setpoint change process and they were conservative. The low level alarm would annunciate before the USAR low level value was reached. The high level alarm would annunciate before the USAR high level value was reached. The licensee also noted that different numbers were used because of the graduations on the gage in the control room. The actual settings needed to be in whole number percentages in this instance.

Basis for NRC Conclusion

The NRC agreed that a violation of 10 CFR 50.71(e) had not occurred primarily because the actual settings were conservative with respect to the USAR values. The NRC noted that the licensee planned to clarify the USAR and this example was not included in the final citation.

10 CFR 50.59(b)(1)

Contested Example 1

USAR, Section X.8.2.8.C, "Common Mode Failure Analysis - Fire," required that combustibles not be located in the service water booster pump room area since both trains of the service water system were located in close proximity. However, Procedure 0.7.1, "Control of Combustibles," Revision 6, allowed up to 90 pounds of wood or 5 gallons of flammable liquid for this area of the plant. In addition, on December 2, 1996, combustible materials (rags, papers, and flammable chemicals) were located in the service water system booster pump room. The safety evaluation for this change in procedure was inadequate, in that a common mode failure analysis had not been performed to justify the presence of combustible materials in the service water booster pump area.

Licensee's View

The licensee noted that the USAR as literally written was unrealistic and had, therefore, been interpreted by plant personnel in a way which was realistic. They stated that a minor level of transient combustibles is always necessary to support maintenance activities. In the past, plant staff had interpreted the absolute requirement that combustibles not be located in the service water booster pump area to really mean no significant level of combustibles should be allowed in the service water booster pump rooms. The licensee also noted that the appropriate level of combustibles had been established during NRC staff review of the Fire Hazards Analysis, which was the basis for Procedure 0.7.1, "Control of Combustibles." The licensee did acknowledge that the USAR should have been updated to accurately describe the requirements for control of combustibles in the service water booster pump area.

Basis for NRC Conclusion

The NRC disagreed with the licensee's conclusions, primarily because the choice to deviate from the absolute requirement that combustibles not be located in the service water booster pump room had not been evaluated. The requirement was originally included in the USAR as the result of a common mode failure analysis. The impact of the addition of the some transient combustibles on the common mode failure analysis had not been evaluated. The NRC included this example in the final citation.