Nebraska Public Power District

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

NLS940119
December 23, 1994

Mr. L. J. Callan
Regional Administrator
NRC Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011

Dear Mr. Callan:

On August 25, 1994, you provided the District with a letter that outlined certain concerns regarding the Cooper Nuclear Station Operations Review Committee (SORC). Your letter stated in part:

> **[O]ur preliminary review has also resulted in our having serious concerns about the functioning of the Station Operations Review Committee (SORC). It is clear from a review of the report [by the Office of Investigations] that members of the SORC did not, in this instance, implement their assigned duties and responsibilities. In addition, it is not apparent that the processes utilized by the SORC functioned to ensure that the SORC's oversight responsibilities were sufficiently independent from outside influences such as senior management and schedular pressures.**

As you are aware, the District has responded in separate correspondence to the NRC regarding alleged outside influences by senior management and alleged schedular pressures on SORC. Accordingly, the following discussion focuses on the functionality of SORC and its ability to implement assigned duties and responsibilities.

Several NRC concerns with SORC have been supported by the District-sponsored September 1, 1994 Diagnostic Self-Assessment Team (DSAT) report and the NRC's November 29, 1994 Special Evaluation Team report. The District agrees with these conclusions and has taken significant steps to improve SORC beginning with an October 5, 1994, management meeting at which the Site Manager provided his expectations regarding SORC performance. He stressed the oversight mission of the SORC, reinforced high standards and expectations, and provided guidance on how the SORC's mission should be completed. The new Plant Manager expressed similar expectations during the October 6, 1994 SORC meeting. More details regarding additional actions taken to achieve these improvements are provided below.

Personnel

SORC membership has changed significantly with the addition of the new Plant Manager (Chairman), the new Engineering Manager, and the new Operations Manager. These personnel have provided a fresh perspective to SORC and provide the appropriate higher safety standards required for continued performance improvement at CNS.

Independent Assessments

In September 1994, a group was convened to review how other utilities conduct the onsite review function, including membership, procedures, and meeting methods. Recommendations from this group included:

- Revise the controlling procedure to eliminate the identification of membership by specific title to provide greater flexibility in establishing the SORC membership.
- Implement a "qualified reviewer" process and/or the use of subcommittees to review and identify items requiring SORC review.
- Upgrade meeting minutes documentation.
- Clearly establish expectations for committee membership and for items being presented for SORC review.

These recommendations were accepted and corrective actions taken. They have led to improved efficiency and quality of SORC assessments.

Also in September 1994, the District brought onsite a recognized authority in the area of nuclear performance assessment and independent oversight improvement to serve as a mentor and coach to the SORC. The initial task of the mentor, however, was to synthesize and evaluate information contained in the DSAT report pertaining to independent assessment activities, including the SORC. This task was performed in conjunction with the external members of the Safety Review and Audit Board (SRAB). During this effort, several weaknesses were revealed regarding independent assessment. These matters have been addressed through the SORC-specific issues that are addressed in the Phase 1 Action Plan Issue closeout (Item 1.2).

On September 23, 1994, CNS Quality Assurance (QA) issued an assessment report to the site manager on SORC. In sum, the QA concerns and recommendations were consistent with conclusions reached by other assessment organizations. On October 27, 1994, the Plant Manager responded to QA by outlining actions that would be, or had been taken, on the stated concerns. QA has since determined that the Plant Manager's actions are acceptable.

Mentorship

Since mid-September, the SORC mentor/coach has worked with CNS line managers to enhance the effectiveness of oversight meetings through various activities, including frequent attendance at meetings, commentary on meeting proceedings and content, promulgation of oversight expectations, individual coaching and feedback for the SORC members, input into Procedure 0.3 revisions, and assumption of a lead role in the development of a training course for the SORC presenters. Also, the mentor/coach has worked closely with new managers throughout their transition at CNS.

Procedure Modification

On November 3, 1994, Procedure 0.3, "Station Operations Review Committee" was revised to describe SORC activities more accurately. Changes included the following.

- Based SORC membership on disciplines (as described in Technical Specifications). Previously, SORC membership was based on position titles.
- Utilization of a matrix format to identify primary and alternate members of the SORC.
- SORC's primary responsibility was clarified to focus on issues relevant to nuclear safety, and to ensure that nuclear safety implications are recognized and properly addressed.

Training

A training course for SORC members and alternates was presented to address the fundamentals of nuclear safety concepts and culture. The training program consisted of two full days in the classroom and covered all aspects of nuclear safety from fundamental philosophy to the bases for design and licensing. In addition, performance-based concepts and evaluation techniques were presented in the training to provide essential skills for applying critical, results-oriented thinking and evaluation techniques for technical, administrative, and organizational problems. Also, the course led to improved insight by SORC members of potential safety impacts of reviewed information.

Conclusion

The District recognizes that the quality of the function that SORC performs is dependent on the attitude of its members, as well as, controlling procedures and processes. Procedures and processes can be changed, but the key ingredient to the success of SORC is the ability to

differentiate between line-management functions and those of independent assessment. Consistent with this approach, new SORC personnel already have improved SORC effectiveness. Items are being presented in more effective ways, and deliberations are clearly focused on nuclear safety. Improved processes and procedures have led to increased efficiency. The actions described in this letter also have resulted in the appropriate intrusive attitude regarding issues being discussed.

As a final note, the District notes its awareness that it must ensure that SORC members have the proper priority between SORC activities and routine job activities. Without this protocol, unacceptable backlogs in SORC issues or work activities could result. The combined actions discussed in this letter respond to NRC concerns. More importantly, these actions respond to what District management believes is necessary for a successfully functioning SORC. Based on the actions that we have taken, the District concludes that all restart issues pertaining to SORC have been resolved.

Sincerely,

Guy Horn

G. R. Horn
Vice-President, Nuclear

cc: U. S. Nuclear Regulatory Commission
Attention: Document Control Desk

NRC Resident Inspector Office
Cooper Nuclear Station

NPG Distribution

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 9, 1995

MEMORANDUM TO: Files

FROM: James R. Hall, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV

James R. Hall

SUBJECT: CLOSURE OF NRR COOPER RESTART ITEMS

The purpose of this memorandum is to document the administrative closure of several items assigned to NRR in the Cooper Restart Action Plan, dated December 21, 1994. These items are identified in the Action Plan Case Specific Checklists, Parts I and II, as specified below.

Case Specific Checklist - Part I:

B.2.a - Issue Daily and Director's Highlights

Copies of all NRR Director's Highlights on Cooper since 8/1/94 were sent to Region IV/DRP (T. Reis) for inclusion in the restart file on 2/3/95.

B.4.5.b - NRC Evaluation of Applicable Items from Section C "ISSUES" Complete

This memorandum documents NRR completion of those items on Part II of the Case Specific Checklist, as discussed below.

B.4.5.f - Comments From Other Parties Considered

A January 20, 1995, note to file from Daniel M. Barss, NRR, documented his discussions with FEMA officials regarding the proposed restart of the Cooper Nuclear Station. Neither FEMA Headquarters nor FEMA Region VII personnel had any offsite emergency preparedness concerns that would affect the proposed restart of Cooper. By letter dated January 25, 1995, Mr. John A. Miller, FEMA Region VII Director, notified the State of Nebraska that the deficiency identified in the November 16, 1994, emergency exercise at Cooper had been corrected.

B.4.6.1 - Applicable License Amendments Have Been Issued

On February 3, 1995, the staff issued Amendment No. 168 for Cooper, revising the Technical Specification definition for Limiting Conditions for Operation. All amendments necessary to support restart have now been issued.

B.4.6.2 - Applicable Exemptions Have Been Granted

Based on our review of the licensee's letter dated January 26, 1995, NRR concludes that no specific exemptions are required prior to allowing

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Cooper restart. NRR is continuing to review certain generic issues that may be applicable to Cooper; those issues will be pursued when the staff's positions are determined.

B.4.6.3 - Applicable Reliefs Have Been Granted

On February 7, 1995, NRR granted a relief request to allow the HPCI turbine exhaust branch line weld to remain in service for the remainder of the current 10-year ISI interval (through the next refueling outage). No other reliefs are required prior to restart.

B.5.c - No Restart Objections From Other Applicable HQ Offices

The Cooper Restart Panel believes that all significant enforcement issues have been sufficiently resolved to allow restart; however, NRR asked the Office of Enforcement if OE had any objection to restart (R. Hall conversation with J. Beall, 2/3/95; no response received). In addition, OCA was notified of Cooper's proposed restart date (R. Hall conversation with T. Madden, 1/31/95; no comments received).

B.5.d - No Restart Objections from Applicable Federal Agencies

FEMA was the only "applicable" Federal agency identified by the Restart Panel for consultation concerning Cooper restart; FEMA expressed no objections, as described in item B.4.5.f above.

B.5.h - Conduct ACRS Briefing/Notification

NRR contacted ACRS staff in December 1994, to determine if a briefing was desired. On February 1, 1995, ACRS staff indicated that the ACRS does not normally request a briefing on plants shut down for less than one year; the ACRS staff will notify NRR if a briefing is desired at some point after Cooper restart (D. Coe, ACRS, Email to R. Hall).

B.5.1.1 - Coordination with Interested Agencies/Parties - FEMA

See item B.4.5.f above.

Case Specific Checklist - Part II:

Item 10.e - Modification to Add REC Containment Isolation Valves

This issue was identified by the licensee in their Operating Experience Review of NRC IN 89-55, which addressed the potential vulnerability of REC piping to a HELB, and the subsequent potential for loss of containment isolation capability. Based on discussions with NRR/SCSB (R. Barrett, J. Pulsipher, W. Long) it appears that the scenario for IN 89-55 is beyond Cooper's licensing basis. The licensee has developed an internal position paper based on a probabilistic safety assessment

(PSA), which concludes that modifications are not required for compliance with the license and would not result in a significant benefit to safety.

On May 24, 1994, Region III submitted a TIA to NRR addressing this same issue for the Dresden station. That TIA requested an NRR position for this issue, which appears to be generic for several Mark I BWRs, including Brown's Ferry. NRR is currently reviewing the TIA and the final staff position will be evaluated for applicability to Cooper. Based on this information, this item is not considered a startup issue for Cooper.

Item 12 - Exceeding Cooldown Rate

This item involved a review of LER 94-015, Revision 1, which described several instances where limits on reactor vessel heatup and cooldown rates were exceeded. NRR has reviewed the licensee's detailed analyses of these events (licensee restart item No. 7.3), and has found the analyses acceptable, as documented in a memorandum from J. Strosnider to W. Beckner, dated February 6, 1995.

Item 13 - FEMA Concern

The identified FEMA deficiency from the November 16, 1994, emergency exercise at Cooper has been adequately resolved, as discussed in the January 25, 1995, letter from John A. Miller, FEMA Region VII Director, to Richard Semm, Nebraska State Civil Defense Agency (See also Item B.4.5.f above).

Docket No. 50-298

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 12, 1994

EA Nos. 94-164, 94-165, 94-166

Nebraska Public Power District
ATTN: Guy R. Horn
Vice President - Nuclear
Post Office Box 499
Columbus, Nebraska 68602-0499

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTIES - $300,000
(NRC Inspection Report Nos. 50-298/94-14, 50-298/94-16, 50-298/94-19)

This refers to three inspections conducted from May 23, 1994, to August 12, 1994, at the Cooper Nuclear Station (CNS) facility, near Brownville, Nebraska. These inspections were conducted specifically to review the circumstances involving: 1) the identification, in May and June, 1994, by the CNS staff of numerous primary containment penetrations that were installed in configurations that were contrary to NRC requirements and never had been local leak rate tested; 2) the circumstances surrounding the declaration of a Notification of Unusual Event and plant shutdown on May 25, 1994, because both emergency diesel generators were declared inoperable; and 3) the identification by the CNS staff of numerous hardware deficiencies that resulted in the failure of the control room envelope pressurization test on April 11, 1994. The reports documenting the results of these inspections were sent to you on September 2, 1994 (50-298/94-14), September 12, 1994 (50-298/94-16), and September 9, 1994 (50-298/94-19). On September 16, 1994, you and other Nebraska Public Power District (NPPD) representatives attended an enforcement conference, which was open to public observation in accordance with the Commissions's continuing trial program, in the NRC's Region IV office to discuss NRC's preliminary conclusion that significant violations of NRC requirements and plant Technical Specifications (TS) had occurred.

On the basis of the information developed during the inspections and the information provided during the enforcement conference, the NRC has determined that a number of significant violations of NRC requirements occurred. Enclosure 1 contains a Notice of Violation and Proposed Imposition of Civil Penalties (Notice) that describes these violations which have been grouped into the following three problem areas, each problem being associated with the safety-related system that was affected:

> Problem I.A consists of violations related to the primary containment system and the failure to maintain the system operable in accordance with the requirements of the plant TS, the failure to adequately test the system in accordance with the requirements of 10 CFR Part 50, Appendix B, Criterion XI, and the failure to maintain an adequate design control of the system in accordance with the requirements of 10 CFR Part 50, Appendix B, Criterion III. These failures to test and maintain the

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primary containment system resulted in containment integrity not being maintained at all times when required because the surveillance requirements for the local leak rate testing of 82 components had never been implemented within the specified surveillance frequency and, when testing was eventually performed, containment leak rate limits were exceeded.

Problem I.B consists of violations associated with the 480 volt and 4160 volt critical buses and the failure to maintain the systems operable in accordance with the requirements of the plant TS and the failure to adequately test the systems in accordance with the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Specifically, CNS had failed to test undervoltage relays associated with the 480 volt and 4160 volt critical buses and because of these failures, the buses have not been capable of performing their safety-related function since operation of the station commenced January 18, 1974.

Problem I.C consists of violations pertaining to the control room emergency filter system and the failure to maintain the system operable in accordance with the requirements of the plant TS and the failure to adequately test the system in accordance with the requirements of 10 CFR Part 50, Appendix B, Criterion XI. As a result of these failures, from June 1989 until April 28, 1994, the control room emergency filter system was not operable at all times when required because testing failed to demonstrate that a positive pressure could be maintained in the control room during the periodic performance of the control room envelope pressurization test.

At the September 16 enforcement conference, NPPD characterized the violations associated with the operability of these systems as having minimal safety significance but acknowledged the regulatory significance of the violations, both individually and collectively. However, it is NRC's view that all of the identified violations associated with the three problems represent significant safety issues. Further, NRC agrees that the violations are of regulatory significance and provide a sufficient basis for the issuance of the three Severity Level III problems.

The NRC believes that chronic and fundamental weaknesses have negatively affected the safety performance of the CNS facility for an unacceptablely long period of time. The description and bases for our concerns with these problems are contained in Enclosure 2.

The results of NPPD's and NRC's inspections into the causes of these violations confirm the NRC's concerns about the effectiveness of NPPD's management of and processes for ensuring that safety systems and components are sufficiently maintained, tested, and controlled such that these systems and components will perform as intended if called upon to mitigate the consequences of an accident. NPPD's and NRC's inspections determined that the following inadequacies caused or contributed to the occurrence of these violations: the continuing practice of inappropriately preconditioning plant hardware to obtain satisfactory test results, which was previously identified

by the NRC in 1993; a continuing failure by NPPD to adequately address identified hardware deficiencies without NRC involvement and prompting; insufficient management oversight, guidance, and monitoring relative to test and design control programs; a failure to identify original design errors; inadequate maintenance and surveillance procedures; and inadequate post-maintenance testing.

During the September 16 enforcement conference, NPPD acknowledged the need for strong, broad-based corrective actions in addition to correcting the specific violations. NRC agrees that broad-based corrective actions are essential. However, the violations that are the subject of this action are similar to violations identified by NRC as early as 1992, and the corrective actions taken to resolve those past problems have been less than fully effective in preventing the recurrence of similar problems. While NRC does not take issue with any of the licensee-identified causes of the violations, NRC believes that the root causes of these violations are more fundamental. Specifically, these violations are indicative of the long-term failure of senior NPPD managers to: 1) implement effective safety processes and procedures; 2) institute a positive station-wide attitude towards identifying and correcting problems; 3) provide effective oversight and monitoring of the CNS staff and programs in order to ensure a high level of safety performance at CNS; and 4) instill and maintain an attitude among plant staff that emphasizes plant safety.

We understand that the actions taken to correct these problems include the following: 1) for Problem I.A, you have taken steps to resolve the design and testing problems and established a program owner and basis document to comply with the requirements of Appendix J of 10 CFR Part 50; 2) for Problem I.B, you have identified and tested all of the affected critical circuits, evaluated and/or tested the remaining circuits, and you have completed, or are in the process of completing a multi-discipline review of your Operational Experience Review Program; 3) for Problem I.C, you installed hardware modifications; and 4) as to each of these problems, you initiated safety culture improvements, revised testing procedures, and expedited design basis reconstitution. In addition, further corrective steps that NRC has required to be taken are outlined in Confirmatory Action Letters 4-94-06, 4-94-06A, 4-94-06B, and 4-94-08. These steps include defining the design basis for your surveillance testing program such that each structure, system, or component is properly tested to validate that it can perform its intended design function, confirming that all discrepancies associated with the primary containment system have been resolved, describing the actions taken to prevent recurrence of the installation of devices that prevent the actuation of safety-related systems, and confirming that inservice inspection of penetration welds are adequately implemented.

Nevertheless, to clearly emphasize the need for senior NPPD managers to identify and undertake sustained actions to improve the overall level of safety performance at CNS, I have been authorized, after consultation with the Commission, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalties (Notice) in the amount of $300,000 for the three Severity Level III problems described above, in Enclosure 2, and in the Notice.

The base value of a civil penalty for a Severity Level III problem is $50,000. Normal application of the enforcement policy could have resulted in significantly higher civil penalty amounts for each of the three Severity Level III problems cited in the Notice. For example, 100% escalation for each problem would be warranted for each of the following factors: 1) past performance, as exhibited by NPPD's poor performance in the areas of problem identification and resolution, maintenance, surveillance testing, and engineering; 2) prior opportunity to identify or prevent the violations because NPPD had multiple prior opportunities to identify and prevent the violations composing Problems I.A., I.B., and I.C.; and 3) duration of the violations, because the violations composing Problems I.A., I.B., and I.C. have existed for many years. Based on these three factors alone, the adjustment factors could have been applied to increase the proposed civil penalty by 300% for each of the problems. However, in light of your initiative to shut down the plant until you could successfully implement an extensive improvement program to address the underlying root causes of performance deficiencies, your commitment not to restart the plant without prior NRC approval, and your significant changes in management oversight of licensed activities, the NRC has decided to exercise broad discretion, pursuant to paragraph VII.B.6. of the Enforcement Policy, and set the civil penalty at $100,000 for each of the Severity Level III problems. Therefore, the total proposed penalty for this action is $300,000.

In addition to the violations discussed above, the Notice contains two Severity Level IV violations which have not been assessed a civil penalty. These violations involve: 1) the failure to update the safety-related drawing list in accordance with an engineering procedure; and 2) the failure to maintain an appropriate procedure for the preventive maintenance of 480-volt circuit breakers.

NPPD is required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing its response. In its response, NPPD should document the specific actions taken and any additional actions it plans to prevent recurrence. After reviewing the response to this Notice, including proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room (PDR). To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. However, if you find it necessary to include such information you should clearly indicate the specific information that you desire not to be placed in the PDR, and, provide the legal basis to support your request for withholding the information from the public.

The responses directed by this letter and the enclosed Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, Pub. L. No. 96-511.

Sincerely,

James L. Milhoan

James L. Milhoan
Deputy Executive Director for
Nuclear Reactor Regulation, Regional
Operations, and Research

Docket No. 50-298
License No. DPR-46
EAs 94-164, 94-165, 94-166

Enclosures: 1. Notice of Violation and Proposed Imposition of Civil Penalties
2. Description of the Severity Level III Problems

Enclosure 1

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTIES

Nebraska Public Power District
Cooper Nuclear Station

Docket No. 50-298
License No. DPR-46
EA Nos. 94-164, 94-165, 94-166

During NRC inspections conducted from May 23, 1994, to August 12, 1994, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the Nuclear Regulatory Commission proposes to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalties are set forth below:

I. Violations Assessed A Civil Penalty

A. 1. Technical Specification 3.7.A.2.a, "Containment Integrity," states, in part, that "primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and when fuel is in the reactor vessel . . . ."

Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," states, in part, that ". . . local leak rate tests (LLRTs) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals no greater than two years. . . . The total acceptable leakage rate for all valves and penetrations other than the MSIVs [main steam isolation valves] is 0.60 La."

Technical Specification 1.Y, "Surveillance Frequency," states, in part, that "performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO [limiting condition for operation] unless otherwise required by the specification."

Contrary to the above, from January 18, 1974, until May 27, 1994, primary containment integrity was not maintained at all times when the reactor was critical or when the reactor water temperature was above 212°F and fuel was in the reactor vessel in that the Surveillance Requirement for the local leak rate testing of 82 components had never been implemented at an interval not to exceed two years. As the result of testing conducted on June 23, 1994, Isolation Valve IA-65CV (one of the 82 components) failed the LLRT, resulting in a total leakage value that significantly exceeded the 0.60 La limit. The 0.60 La limit corresponds to a leakage rate of 5.37 scmh (189.60 scfh). The LLRT

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failure of Valve IA-65CV resulted in a total leakage rate that exceeded 17.66 scmh (623.57 scfh). (01013)

2. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents."

   CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11, written to implement the requirements of 10 CFR Part 50, Appendix B, Criterion XI, requires that each type of test program performed will be defined by written procedures and instructions, and it requires that acceptance tests will be developed for structures, systems, and components to demonstrate their capability to perform satisfactorily following repairs or modification.

   a. Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. Specifically as of May 14, 1994, 68 components passing through 54 primary containment penetrations, each required to be local leak rate tested in accordance with the requirements of Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," had not been identified in a procedure as requiring local leak rate testing, as required by CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11. These components had never been local leak rate tested. (01023)

   b. Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. As of June 21, 1994, instrument pressure switches PC-PS-12A, B, C, and D; PC-PS-101A, B, C, and D; PC-PS-119A, B, C, and D; PC-PS-16; and PC-PT-512A and B, each required to be local leak rate tested in accordance with the requirements of Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," had not been identified in a procedure as requiring local leak rate testing after being isolated from the containment integrated leak

rate test, as required by CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11. These switches had never been local leak rate tested. (01033)

3. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states in part, that "[m]easures shall be established to assure that . . . the design basis . . . [is] correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

   Draft General Design Criterion 53, a measure written to comply with the requirements of 10 CFR Part 50, Appendix B, Criterion III, as committed to in Appendix F of the Updated Safety Analysis Report (USAR), states that "[p]enetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus."

   Draft General Design Criterion 1, as committed to in Appendix F of the USAR, states that "[t]hose systems and components of reactor facilities which are essential to the prevention of accidents which could effect the public health and safety or mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed."

   General Electric Design Specification No. 22A1153, "Codes and Industrial Standard," Revision 1, states, in Note 3 of the Appendix, that "[p]iping, which is an integral part of the primary containment for isolation purposes, shall have at least the same quality and levels of assurance as the primary containment."

   Contrary to Criterion III, the licensee did not assure that the above design bases were correctly translated into specifications and instructions and did not assure that deviations from quality standards were controlled. Specifically:

   a. As of May 14, 1994, numerous primary containment penetrations had no redundant valving. These penetrations included, but were not limited to, Penetrations X-21, X-22, X-25, X-29E, X-30E/F, X-33E/F, X-209A/B/C/D, and X-218.

   b. As of February 22, 1994, 10 penetrations consisting of manually operated vents, drains, and test connections

and requiring closure for the containment function, had a single manual isolation valve for containment isolation as opposed to the required redundant valving and associated apparatus.

c. During an NRC inspection conducted June 13, 1994, through August 12, 1994, it was determined that approximately 300 containment penetrations had not been designed, fabricated, or installed to the same standards as the primary containment because these components had not been correctly classified as essential. As a result, these penetrations had not been designed, fabricated, or erected to quality standards that reflect the importance of the safety function to be performed (e.g., some welds were not nondestructively tested, some penetrations were not local leak rate tested, and the penetrations were not treated as safety- related by the licensee's quality assurance program). (01043)

This is a Severity Level III problem (Supplement I).
Civil Penalty - $100,000

B. 1. Technical Specification 3.9.A.1.c, "Auxiliary Electrical Equipment," requires, in part, that the reactor shall not be made critical from a Cold Shutdown Condition unless the 4160 volt critical buses 1F and 1G and the 480 volt critical buses 1F and 1G are energized, and the undervoltage and loss of voltage relays, as well as their auxiliary relays, are operable.

Technical Specification Surveillance Requirement 4.9.A.1.a, "Emergency Buses Undervoltage relays," states that "once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators." USAR Section 2.2.7.2.1.a, "Standby A-C Power (Diesel Generators) Test Capability," defines the function of the protective scheme as providing for the clearing the buses of all motor loads excepting supply to the 480 volt critical unit substation.

Technical Specification 1.Y, "Surveillance Frequency," states, in part, that "performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO [limiting condition for operation] unless otherwise required by the specification."

Contrary to above, from January 18, 1974 until May 25, 1994, the reactor had been made critical without 4160 volt

critical buses 1F and 1G, and 480 volt critical buses 1F and 1G being operable in that the undervoltage relays associated with several of the electrical loads supplied by these buses had never been tested to demonstrate their operability or, upon testing, failed to perform their intended function of shedding their respective electrical loads from these buses. (02013)

2. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptable limits contained in applicable design documents."

   Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. Specifically:

   a. During an NRC inspection conducted May 23, 1994, through August 12, 1994, Procedure 6.3.4.3, "Sequential Loading of Emergency Diesel Generators," Revision 31, which is performed to satisfy Technical Specification Surveillance Requirement 4.9.A.1.a, "Loss of Voltage Relays," was determined to be inadequate because it did not assure that the emergency diesel generators and critical buses would perform satisfactorily in service in that the procedure did not contain requirements to verify that the 480-volt supply breakers for safety-related and nonsafety-related loads would shed from their electrical buses within a specified time, nor did the procedure identify that the control rod drive pump motors and station air compressors were required to be shed from the electrical bus.

   b. During an inspection conducted May 23, 1994, through August 12, 1994, the NRC identified that Procedure 6.3.20.1, "RHR Service Water Booster Pump Flow Test and Valve Operability Test," Revision 27, did not provide for the testing of the load shedding feature of the supply breakers associated with the 4160 volt residual heat removal service water booster pumps. (02023)

This is a Severity Level III problem (Supplement I).
Civil Penalty - $100,000.

C. 1. Technical Specification 3.12.A.1, "Control Room Emergency Filter System," states, in part, that ". . . the Control Room Emergency Filter system . . . shall be operable at all times when containment integrity is required."

The Order Confirming Licensee Commitments on Post-TMI Related Issues, dated July 10, 1981, confirms NPPD's commitment to complete NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability." Item III.D.3.4 involves the review of facility design requirements against the Standard Review Plan. The NPPD response to Generic Letter 80-90, dated December 30, 1980, submitted the control room habitability evaluation, which stated, in part, "the CNS control room ventilation system is designed to maintain the control room at about ¼-in. H_2O [0.031 kPa] positive pressure by supplying air at a high enough pressure that even when system losses and the booster exhaust fan pressures are accounted for, the control room pressure is still positive. . . "

A Safety Evaluation Report for the Cooper Station from the Accident Evaluation Branch on NUREG-0737, Item No. III.D.3.4, "Control Room Habitability," dated February 24, 1982, states, in part, that ". . . the design meets the criteria identified in Item III.D.3.4 of NUREG-0737 and is acceptable."

Contrary to the above, from June 1989 until April 28, 1994, the Control Room Emergency Filter system was not operable at all times when containment integrity was required in that testing failed to demonstrate that a positive pressure could be maintained in the control room during the periodic performance of the control room envelope pressurization test. (03013)

2. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents."

CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11, written to implement the requirements of 10 CFR Part 50, Appendix B, Criterion XI, requires that each type of test program performed will be defined by written procedures and instructions, and it requires that acceptance tests will be developed for

structures, systems, and components to demonstrate their capability to perform satisfactorily following repairs or modification.

Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits in applicable design documents. Specifically, from June 1989 until June 1994, Surveillance Procedure 6.3.17.18, "Control Room Envelope Pressurization Test," Revision 4, was not sufficiently detailed in that it did not incorporate acceptance limits to assure that the Control Room Emergency Filter system would perform satisfactorily in service and because the procedure did not prohibit the inappropriate manipulation of pressures in the adjoining buildings as a precondition for conducting the test. (03023)

This is a Severity Level III problem (Supplement I).
Civil Penalty - $100,000.

II. <u>Violations Not Assessed A Civil Penalty</u>

A. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," states, in part, that "[a]ctivities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings."

Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7, written, in part, to implement 10 CFR Part 50, Appendix B, Criterion V, requires that safety-related drawings be included on the safety-related drawing list.

1. Contrary to the above, during an NRC inspection conducted June 13, 1994, through August 12, 1994, it was determined that safety-related Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," Revision N27, was not included on the safety-related drawing list. As a result of this determination, the licensee subsequently identified 13 other drawings containing safety-related components that were not included on the safety-related drawing list. (04014)

This is Severity Level IV violation (Supplement I).

2. Contrary to the above, during an NRC inspection conducted May 23, 1994, through August 12, 1994, Maintenance Procedure 7.3.2.1, "DB-25 and DB-50 Circuit Breakers - Setting, Testing, and Maintenance (With Amptectors)," Revision 3, was

determined to be inappropriate to the circumstances in that the procedure did not contain a requirement to remove tie-wraps from the subject circuit breakers following preventive maintenance, nor did the procedure provide for comprehensive postmaintenance testing of all circuit breaker functions following the completion of preventive maintenance. (05014)

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalties (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each alleged violation: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. 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. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalties by letter addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, with a check, draft, money order, or electronic transfer payable to the Treasurer of the United States in the amount of the civil penalties proposed above, or may protest imposition of the civil penalties in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalties will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalties, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violation(s) listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties in whole or in part, such answer may request remission or mitigation of the penalties.

In requesting mitigation of the proposed penalties, the factors addressed in Section VI.B.2 of 10 CFR Part 2, Appendix C, should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the

Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing civil penalties.

Upon failure to pay any civil penalties due which subsequently have been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234(c) of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, letter with payment of civil penalties, and Answer to a Notice of Violation) should be addressed to: Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555 with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas, 76011 and a copy to the NRC Resident Inspector at the facility that is the subject of this Notice.

Dated at Rockville, Maryland
this 12th day of December, 1994.

Enclosure 2 - Description of the Severity Level III Problems

A. <u>Primary Containment System Problem (I.A)</u>

From NRC's perspective, the three violations associated with the primary containment system problem were important from a safety and regulatory significance standpoint because NPPD's surveillance testing and system design control programs for the primary containment system possessed neither the rigor nor the technical depth to be able to maintain the original design or adequately test the operability of the system. As a result, NPPD was unable to reliably determine system operability, and the consequence of this failure was that the containment system was inoperable from the original licensing in 1974, until the problem was identified by NRC and NPPD personnel in May 1994. These longstanding problems with the design and testing of the primary containment system were permitted to exist for an extended period of time and the failure of NPPD management and staff to identify and correct these problems resulted in the continuing degradation of this safety-related system.

The failure to maintain primary containment integrity (Violation I.A.1) is considered safety significant because, despite NPPD being able to demonstrate by analysis, performed after the violation was identified, that although the primary containment system was able to perform its intended safety function, the leakage rate associated only with isolation valve IA-65CV greatly exceeded the combined maximum permissible leakage rate for all primary containment penetrations. In arriving at its conclusion that the primary containment system was capable of performing its intended safety function, NPPD assumed a failure mode of the nonsafety-related instrument air system such that all the post-accident primary containment bypass leakage would still be filtered, resulting in control room and off-site dose rates that did not exceed regulatory requirements. Since NPPD did not assume an unfiltered release relative to this analysis, the potential for excessive control room and off-site dose rates was not calculated and evaluated for significance.

Violation I.A.2 is discussed in Section D of this enclosure.

Violation I.A.3 associated with the primary containment system problem consists of a systematic failure in the design control of the containment system and is considered equally safety-significant. NRC recognizes that the primary containment penetration design deficiencies occurred during the construction of the plant, and NRC also recognizes that these deficiencies were identified as a result of implementing corrective actions for violations identified by NRC in 1993. However, the corrective actions that resulted in the identification of these deficiencies were <u>not</u> implemented in response to previous <u>escalated</u> enforcement action taken in EA 93-137. Additionally, NRC believes that the primary containment penetration design deficiencies should have been identified by routine efforts much sooner. Therefore, discretion or mitigation for this violation as discussed in Sections VII.B.4 and 5 of the Enforcement Policy is not appropriate.

B. Critical Buses Problem (I.B)

The inoperability of the 480 volt and 4160 volt critical buses (Violation I.B.1) is also considered significant from a safety and regulatory standpoint. At the September 16 enforcement conference, NPPD presented the results of an evaluation of emergency diesel generator loading, which assumed that shedding of non-essential loads would not occur. On the basis of this evaluation, NPPD concluded that the emergency diesel generators would still be capable of performing their intended safety function. The NRC review of this evaluation is still ongoing; however, based on a preliminary review, NRC has no basis for disputing the conclusions reached by NPPD with regard to emergency diesel generator operability, even though the NRC did identify several weaknesses in the NPPD evaluation.

Although the emergency diesel generators appear to have remained functional during the period the undervoltage relays associated with certain electrical loads were not tested or were not operable, NRC has concluded that the critical buses 1F and 1G (480 volt and 4160 volt), which are supplied with electrical power by the emergency diesel generators under certain accident conditions, were inoperable. This conclusion was based on the undervoltage relays associated with some of the electrical loads that are supplied by these buses being inoperable. NRC's determination that undervoltage relays were inoperable was based on the relays: 1) having never been tested (e.g., station air compressors 1A and 1B which are supplied by 480 volt critical buses 1F and 1G); 2) failing to function as intended during testing (e.g., motor control centers N, V, MR, and OG-1 which are also supplied by 480 volt critical buses 1F and 1G); and 3) failing to function as the result of configuration changes (e.g., residual heat removal service water booster pumps 1A-D which are supplied by 4160 volt critical buses 1F and 1G). Consequently, NRC has recharacterized the apparent violation of TS 3.9.A.1.b (298/9416-03) as a violation of TS 3.9.A.1.c. TS 3.9.A.1.c requires that the reactor not be made critical from a Cold Shutdown Condition unless the 4160 volt critical buses 1F and 1G and the 480 volt critical buses 1F and 1G are energized, and the undervoltage and loss of voltage relays, as well as their auxiliary relays, are operable.

Violation I.B.2 is discussed in Section D of this enclosure.

C. Control Room Emergency Filter System Problem (I.C)

The circumstances involving the control room emergency filter system problem consisted both of inadequacy of the test control program and violations of the plant TS (Problem I.C.1), and illustrate a serious NRC concern with the maintenance and testing practices at CNS. NRC identified inadequacies in these areas in 1993 relative to the testing and maintenance of the secondary containment. The corrective actions for those problems, however, did not result in the identification of the inoperable control room emergency filter system. The continuing practice of inappropriate preconditioning of structures, systems, and components to obtain satisfactory test results, as evidenced by the

manipulation of the pressures in the adjoining buildings to obtain the required pressure in the control room during the control room envelope pressurization test, has masked hardware deficiencies which have caused or contributed to this inoperable safety system.

Violation I.C.2 is discussed in the following section of this enclosure.

D. Testing Safety-Related Systems Violations

In NRC's view, the violations associated with NPPD's failure to adequately test safety-related systems (Violations I.A.2, I.B.2, and I.C.2), a failure common to all three problems, represent deep-seated issues, indicative of a collapse in the test control program at CNS. Test control program failures include: inappropriate preconditioning of structures, systems, and components; long-standing surveillance procedure deficiencies that the CNS staff either did not recognize or had accepted, and resulted in a failure to demonstrate system operability; components that were required to be tested by the plant TS or NRC regulations but were never tested; and a failure by NPPD management to ensure that corrective actions for past similar surveillance-related problems were adequate to prevent recurrence. As a result, NRC is citing test control program violations as a part of each of the three identified problems.

Nebraska Public Power District

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 40C
ARLINGTON, TEXAS 76011-8064

SEP 26 1994

Docket: 50-298
License: DPR-46

Nebraska Public Power District
ATTN: Guy R. Horn, Vice President - Nuclear
P.O. Box 499
Columbus, Nebraska 68602-0499

SUBJECT: ENFORCEMENT CONFERENCE

This refers to the enforcement conference, conducted on September 16, 1994, at the Region IV office in Arlington, Texas, concerning activities authorized by NRC License DRP-46 for the Cooper Nuclear Station. The meeting was attended by those on the attached Attendance List The subjects discussed at this meeting are described in the enclosed Meeting Summary.

It is our opinion that this meeting was beneficial and has provided a better understanding of the apparent violations identified in NRC Inspection Reports 50-298/94-14, -94-16, and -94-19 and your corrective actions. Your staff's professionalism and candor during the meeting was greatly appreciated. Any enforcement actions taken as a result of the meeting will be addressed in separate correspondence.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter will be placed in the NRC's Public Document Room.

Should you have any questions concerning this matter, we will be pleased to discuss them with you.

Sincerely,

A. Bill Beach, Director
Division of Reactor Projects

Attachments:
1. Attendance List
2. Licensee Presentation

cc:
Nebraska Public Power District
ATTN: G. D. Watson, General Counsel
P.O. Box 499
Columbus, Nebraska 68602-0499

~~9410040009~~

Nebraska Public Power District
ATTN: John Mueller, Site Manager
P.O. Box 98
Brownville, Nebraska 68321

Midwest Power
ATTN: James C. Parker, Sr. Engineer
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303

Lincoln Electric System
ATTN: Mr. Ron Stoddard
11th and O Streets
Lincoln, Nebraska 68508

Nebraska Department of Environmental
Quality
ATTN: Randolph Wood, Director
P.O. Box 98922
Lincoln, Nebraska 68509-8922

Nemaha County Board of Commissioners
ATTN: Larry Bohlken, Chairman
Nemaha County Courthouse
1824 N Street
Auburn, Nebraska 68305

Nebraska Department of Health
ATTN: Harold Borchert, Director
Division of Radiological Health
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509-5007

Department of Natural Resources
ATTN: R. A. Kucera, Department Director
of Intergovernmental Cooperation
P.O. Box 176
Jefferson City, Missouri 65102

Kansas Radiation Control Program Director

bcc to DMB (IE45)

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L. J. Callan
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K. Perkins, D:WCFO
C. Hackney, RSLO

Resident Inspector
Leah Tremper, OC/LFDCB, MS: MNBB 4503
DRSS-FIPB
Project Engineer (DRP/C)
Senior Resident Inspector - River Bend
J. Gilliland, PAO

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RCStewart;df	PHHarrell	ABBeach		
9/26/94	9/26/94	9/26/94		

MEETING: ENFORCEMENT CONFERENCE W/ NPPD

SUBJECT: APPARENT INOPERABILITY OF CONTAINMENT, CONTROL ROOM ENVELOPE & EDGS

DATE: SEPTEMBER 16, 1994

ATTENDANCE LIST

(PLEASE PRINT CLEARLY)

NAME	ORGANIZATION	POSITION TITLE
P. HARRELL	NRC	CHIEF, BRANCH C, DRP
T. P. GWYNN	NRC RIV	DIRECTOR, DRS
A. BILL BEACH	NRC RIV	DIRECTOR, DRP
L. JOE CALLAN	NRC RIV	REGIONAL ADMINISTRATOR
JOHN MONTGOMERY	NRC RIV	DEPUTY REGIONAL ADMINISTRATOR
WILLIAM BECKNER	NRC NRR - PDIV-1	DIRECTOR PDIV-1
C. J. Paulk	NRC RIV	Acting Chief, Eng Branch, DRS
R.A. KOPRIVA	NRC RIV	SENIOR RESIDENT INSPECTOR COOPER
J.R. HALL	NRC NRR PM	PROJECT MANAGER - COOPER
GR HORN	NPPD	Vice President, Nuclear
ROBERT GODLEY	NPPD	NUCLEAR LICENSING MANAGER
John Mueller	NPPD	Site Manager
M.T. BOYCE	NPPD	DESIGN BASIS SUPERVISOR
SCOTT S. FREBORG	NPPD	PLANT ENGR. SUPERVISOR
Daniel E. Buman	NPPD	Lead Electrical Engineer
GAUTAM SEN	NPPD	Senior Staff Licensing Engineer
Art Wiese	NPPD	Lead Engr. - Nuc Eng. Dept.
MICHAEL SPENCER	NPPD	ENGINEERING PROGS. SUPERVISOR
JIM GAUSMAN	NPPD	ENGINEERING MANAGER
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Nebraska Public Power District Cooper Nuclear Station
TIME/DATE	September 16, 1994 at 9 a.m.
MEETING LOCATION	Training Conference Room
EA NUMBER	EA 94-164, EA 94-165, EA 94-166

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
NRC ATTENDEES		
Art Howell	Office of RA	Technical Assistant
PAULA GOLDBERG	REGION IV - DRS	REACTOR INSPECTOR
Ann Hodgdon	OGC	Acting Regional Counsel
TERRY REIS	REGION IV - DRP	PROJECT ENG
ROBERT C. STEWART	REG. IV - DRP	(ACTING) PROJECT ENGINEER
RONALD D. STODDARD	LINCOLN ELEC. SYSTEM	CHIEF ENGINEER
RICHARD SINGER	Midwest Power	Manager Nuclear Admin.
MARK A. SATORIUS	OFFICE OF ENFORCEMENT	ENFORCEMENT SPECIALIST
WAYNE C WALKER	REG IV - DRP	RESIDENT INSPECTOR - CNS
JOHN L PELLET	REG IV DRS	Acting Deputy Director

ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Nebraska Public Power District Cooper Nuclear Station
TIME/DATE	September 16, 1994 at 9 a.m.
MEETING LOCATION	Training Conference Room
EA NUMBER	EA 94-164, EA 94-165, EA 94-166

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
	ATTENDEES	
GARY SANBORN	NRC REGION 4	ENF. OFFICER

ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Nebraska Public Power District Cooper Nuclear Station
TIME/DATE	September 16, 1994 at 9 a.m.
MEETING LOCATION	Training Conference Room
EA NUMBER	EA 94-164, EA 94-165, EA 94-166

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
LICENSEE ATTENDEES		
RICHARD A SESSIONS	NEB PUBLIC POWER DISTRICT	DIV MGR QA
RAYMOND G. JONES	NEB PUBLIC POWER DISTRICT	SR MGR SAFETY ASSESSMENT

ENFORCEMENT CONFERENCE ATTENDANCE

LICENSEE/FACILITY	Nebraska Public Power District Cooper Nuclear Station
TIME/DATE	September 16, 1994 at 9 a.m.
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EA NUMBER	EA 94-164, EA 94-165, EA 94-166

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
	ATTENDEES	
Walter A. Canney	Lincoln Elect. System	Administrator

ATTACHMENT 2

NEBRASKA PUBLIC POWER DISTRICT

NUCLEAR REGULATORY COMMISSION - REGION IV

ENFORCEMENT CONFERENCE

NRC INSPECTION REPORT NOS. 50-298/94-14, 16, 19

SEPTEMBER 16, 1994

AGENDA

I.	INTRODUCTION/OVERVIEW	J. MUELLER
II.	DISCUSSION OF ISSUES/ ISSUE-SPECIFIC CORRECTIVE ACTIONS	
	A. CONTROL ROOM ENVELOPE	S. FREBORG
	B. CONTAINMENT PENETRATIONS	M. BOYCE
	C. UNDERVOLTAGE DEVICES	D. BUMAN
III.	CLOSING REMARKS	J. MUELLER

INTRODUCTION/OVERVIEW

J. MUELLER

DISCUSSION OF APPARENT VIOLATION

CONTROL ROOM ENVELOPE

S. FREBORG

CONTROL ROOM EMERGENCY FILTER SYSTEM CHRONOLOGY

April 11, 1994	Control Room Emergency Filter System declared inoperable to perform preventive maintenance on envelope boundary door.
April 11, 1994	Surveillance Procedure 6.3.17.18, "Control Room Envelope Pressurization Test," performed -- test failed - LCO 3.12.A.3 allows continued plant operation for 7 days.
April 11-April 18, 1994	Repair efforts performed; re-test did not pass.
April 18, 1994	NRC grants 14 day extension to LCO 3.12.A.3.
April 18-April 25, 1994	Leak repair efforts conducted.
April 25, 1994	STP 93-257 performed to quantify interaction between the Control Room Envelope and plant areas that share common boundary conditions.
April 28, 1994	S.P. 6.3.17.18 completed satisfactorily.
April 30, 1994	CREFS declared operable.
May 25, 1994	CNS shutdown due to undervoltage device tie-rap issue.

CONTROL ROOM EMERGENCY FILTER SYSTEM CHRONOLOGY

June 22, 1994	S.P. 6.3.17.18 revised to include administrative and operability acceptance criteria.
June 23, 1994	Test performed on Control Room Envelope emergency mode -- failed administrative criteria; satisfied operability criteria; CR initiated.
July 7, 1994	Technical meeting between Nebraska Public Power District and the Nuclear Regulatory Commission Regarding Control Room Emergency Filter System.

CONTROL ROOM ENVFLOPE

APPARENT VIOLATION 94-19-01

Technical Specification 3.12.A.1, "Control Room Emergency Filtration System" (CREFS) was not satisfied in that the CREFS could not meet its intended design function (i.e., maintain a positive pressure). Therefore, the Control Room Emergency Filter System, which provides air flow to maintain a positive pressure in the Control Room Envelope, was inoperable from 1989 to 1994.

DISCUSSION:

- The surveillance procedure was inadequate.
- The maintenance program to maintain the Control Room Envelope was inadequate.
- Lessons learned from secondary containment issue were not adequately applied to the Control Room Envelope.
- Engineering assumptions regarding design basis were nonconservative.
- While licensee personnel manipulation of pressures in adjoining buildings to obtain the required pressure in the Control Room did not violate procedures, it was an unacceptable practice.

CONTROL ROOM ENVELOPE (CONT)

APPARENT VIOLATION 94-19-01

SAFETY SIGNIFICANCE:

- Operator doses would have remained within GDC 19 and Standard Review Plan, Section 6.4 Limits based upon analysis of projected unfiltered inleakage.

REGULATORY SIGNIFICANCE:

- Continuing failure by NPPD to adequately address Control Room Emergency Filter System deficiencies.

MITIGATING FACTORS:

- NPPD identified this deficiency and pursued necessary resolution.

CORRECTIVE ACTIONS:

- Hardware modifications.

- Safety culture improvements.

CONTROL ROOM ENVELOPE (CONT)

CORRECTIVE ACTIONS	STATUS
CAUSE:	
INADEQUATE DESIGN WITH INSUFFICIENT MARGIN.	
ACTION (Short Term):	
Install 1000 CFM capacity emergency fan to increase the pressurization margin within the existing capability of the filter.	Modification to be installed upon approval of required Technical Specification change, which is expected the first week of October.
ACTION (Long Term):	
Implement necessary modifications as a result of the completion of the Control Room Habitability System design criteria document.	Control Room Habitability System Design Criteria Document scheduled for completion 9-30-94. Submit EWR for any necessary modifications by 10-15-94.

CONTROL ROOM ENVELOPE (CONT)

CORRECTIVE ACTIONS	STATUS
<u>CONTRIBUTING FACTORS:</u> • INADEQUATE PM FREQUENCY FOR INSPECTION AND LUBRICATION OF DOOR SEALS. • SENSITIVITY OF EXISTING DESIGN TO PROPER BALANCE BETWEEN CONTROL ROOM AND CABLE SPREADING ROOM NOT RECOGNIZED. • LEAK PATHS CREATED DUE TO INADEQUATE WORK INSTRUCTIONS. • DOORS OPENING IN WRONG DIRECTION (POSITIVE PRESSURE TENDS TO DISENGAGE DOOR FROM DOOR SEAL). • INADEQUATE TURBINE BUILDING DP SENSOR MAINTENANCE.	

CONTROL ROOM ENVELOPE (CONT)

CORRECTIVE ACTIONS	STATUS
ACTIONS:	
• Increase frequency of Control Room envelope door seal PM (07678) to weekly	Complete.
• Revise Procedure 7.0.1/7.0.5 to ensure that all maintenance activities that relate to Control Room envelope doors or penetrations or that affect adjacent area HVAC Systems (listed in S.P. 6.3.17.18) flow rate or balance; include PMT requirements to perform 6.3.17.18.	Complete.

CONTROL ROOM ENVELOPE (CONT)

CORRECTIVE ACTIONS	STATUS
• Record the flow rates for all flow paths in the Control Room Ventilation System during both normal and emergency modes of operation to provide a baseline for future system balancing.	This item is being re-evaluated. The implementation of DC 94-262 may negate the need to implement this corrective action.
• Evaluate if a revision to design review checklists in CNS Procedure 3.4.6 is needed to provide more appropriate questions to ensure that the Control Room pressure boundary requirements are adequately addressed in DCs.	Scheduled for completion 10-31-94.
• Revise 6.3.17.18, as necessary, to control all variables in a manner to ensure consistent test results that are representative of the limiting post-accident conditions.	Complete.
• Evaluate the possibility of caulking the inactive leaf on doors H300, N305 to reduce leakage through the seals.	Complete. Evaluation indicates caulking is unnecessary.
• Increase the frequency of performance of S.P. 6.3.17.18 from once per cycle to monthly.	Complete.

CONTROL ROOM ENVELOPE (CONT)

CORRECTIVE ACTION(S)	STATUS
• Develop a PM to inspect outside air sensing line filters on an annual basis for filter element plugging and gasket leaks. The sensing lines should be blown out at the same time and inspected for leaks or damage.	Complete.
• Evaluate the use of a more accurate manometer for all future S.P. 6.3.17.18 testing. If use of the present manometer is continued, a more rigid mounting method should be determined.	Complete. Evaluation concludes that existing manometer should be retained for S.P. 6.3.17.18.
• Investigate the possibility of replacing door H200 with a door that swings inward.	Complete. Replacement of this door is unnecessary due to the low dP experienced.

CONCLUSION

- Control Room Envelope could not consistently satisfy design function.
- Adequate corrective actions have been taken to address hardware issues.
- Hardware modifications will significantly increase available margin.
- Steps have been taken to improve the safety culture.

DISCUSSION OF APPARENT VIOLATIONS

CONTAINMENT PENETRATIONS

M. BOYCE

CHRONOLOGY OF PENETRATION ISSUES

June 17, 1993	Inspection Report 93-17 issued.
October 12, 1993	Notice of violation and proposed imposition of civil penalty from IR 93-17. Level IV violation for failure to have an appropriate rationale for reverse direction testing. Level IV violation for failure to leak rate test the internals of the hydrogen/oxygen analyzers.
November 12, 1993	District response to IR 93-17 made commitment to conduct a "detailed review of all containment penetrations and their associated Appendix J testing requirements... Prior to startup from the next refueling outage."
December 2, 1993	Memo from R. E. Wilbur to K. C. Walden directing that the design basis reconstitution for the Primary Containment System be expedited.
January 1994	Work started on Primary Containment design basis document.

CHRONOLOGY OF PENETRATION ISSUES

February 1994	Development of Primary Containment walkdown procedure started (SP 94-202).
May 17, 1994	SP 94-202, Primary Containment walkdown procedure approved.
May 18, 1994	SP 94-202 walkdown started.
May 27, 1994	Plant in cold shutdown due to Electrical Distribution System concerns.
June 2, 1994	CR 94-241 written due to discovery of penetration X-218.
June 6, 1994	CR 94-261 written due to discovery of penetration X-209A-D.
June 6, 1994	Started review of as-built containment penetration information.
June 11, 1994	CR 94-286 written documenting numerous design and testing concerns with Primary Containment.

CHRONOLOGY OF PENETRATION ISSUES

June 11 through June 20, 1994	Continued review penetration-by-penetration of containment design and testing.
June 14, 1994	Presentation to NRC Region IV on Primary Containment.
June 20, 1994	CR 94-0325 written to document IIIN and IVP piping penetrating Primary Containment.
June 21, 1994	CR 94-0377 Written to document failure to leak rate test pressure instruments.
June 23, 1994	CR 94-0378 written to document LLRT failure of IA-65CV.
June 27, 1994	Technical presentation to NRR and Region IV in Rockville.
July 29, 1994	NRC Management Meeting.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 1

APPARENT VIOLATION

Containment isolation valves that existed in the plant were not shown on flow diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," revision N27. Flow diagram No. 2028 was not included on the safety-related drawing list in accordance with Cooper Nuclear Station Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7.

NRC identified this as Example 1 of an apparent violation of 10 CFR 50 APPENDIX B, Criterion III.

DISCUSSION:

- The District violated Procedure 3.8 in that the drawing list was incorrect.
  - Did not affect control of safety-related drawings.
  - Reason for list was as-build project scope (2028 should be on list).
  - Procedure 3.7 controls distribution of safety-related drawings.
  - Other parts of Procedure 3.8 control revision.
- There were 5 valves not shown on drawing 2028.
  - 4 Manual instrument air valves to AOVs in drywell.
  - 1 Manual valve on vent line.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 1

SAFETY SIGNIFICANCE:

- No effect on component classification or drawing revision, content, distribution, or use.
  - Procedure 3.13 Controls classification.
- Did not affect treatment of any safety-related component (maintenance, testing, design, etc.).
  - Engineering Date File (EDF) Provides component classification for work/design control.
- Manual vent valve was shown on a drawing but was not shown in the correct configuration.
  - Two valves and cap shown vs. one valve and cap actual.
- The Instrument Air Valves are on other drawings and in EDF and controlled by operating procedures.
- Minimal safety significance.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 1

REGULATORY SIGNIFICANCE:

- Had 2028 been on the list, it would have been in as-built verification project.
- Should have recognized and corrected the deficiency in Procedure 3.8.

MITIGATING FACTORS:

- The District identified the drawing errors during the containment walkdown.
- There was no impact on safety-related drawing control due to the missing drawing on the Procedure 3.8 list.

CAUSE:

- Insufficient management oversight and guidance.
  - Recognized drawing list discrepancies and failed to take corrective actions.
  - Failed to establish an adequate program to identify correct drawing scope.

CORRECTIVE ACTIONS:

- Revising Procedure 3.8.

CONTAINMENT PENETRATIONS

APPARENT VIOLATION 1, EXAMPLE 1

CORRECTIVE ACTION(S)	STATUS
CAUSE: INSUFFICIENT MANAGEMENT OVERSIGHT AND GUIDANCE. ACTIONS: Revising Procedure 3.8.	 Procedure change in progress.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 2

APPARENT VIOLATION:

The Licensee identified a number of examples where penetrations were found to lack redundant containment isolation.

The failure to have redundant containment isolation barriers was identified as an apparent violation of 10 CFR 50, Appendix B, Criterion III.

DISCUSSION:

- 13 Missing caps on vents, drains or test connections.
- 30 single manual valves on process lines.
- 5 thermocouples through piping and a disabled open valve with sealant compound inside.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 2

SAFETY SIGNIFICANCE:

- MISSING CAPS
  - Each of the 13 cases of missing caps had a normally closed manual valve in the line.
  - Each of the manual valves had either been tested during the ILRT or would not be exposed to containment atmosphere.
- SINGLE VALVES
  - Each of the 30 single manual valves either was not in direct communication with the containment atmosphere or was ILRT tested.
  - As-found LLRTs successful.
- THERMOCOUPLES
  - Subjected to ILRT pressure.
  - Passed an as-found LLRT (cumulative leakage 0.11 SCFH).
  - Passive barriers.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 2

REGULATORY SIGNIFICANCE:

- Result of original design and should have been identified sooner.
- Lack of appropriate procedural controls on manual valves and caps.

MITIGATING FACTORS:

- Identified as a result of broad corrective actions for IR 93-17 violation.
- Single manual valve isolation resulted from the DISTRICT'S original Licensing position on Safety Guide 11 as provided in FSAR Question 5.5 and as included in the FSAR. The DISTRICT does not consider this an acceptable position to meet the containment design objectives.
- Original plant design, not design change process.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 2

CAUSE:

- Failure to look beyond the requirements of the USAR to identify the appropriate design requirements for these containment penetrations.

CORRECTIVE ACTION:

- Review each penetration to ensure safety function can be achieved and resolve discrepancies.
- Expedite design basis reconstitution.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 2

CORRECTIVE ACTION(S)	STATUS
CAUSES: FAILURE TO LOOK BEYOND THE REQUIREMENTS OF THE USAR TO IDENTIFY THE APPROPRIATE DESIGN REQUIREMENTS FOR THESE CONTAINMENT PENETRATIONS. ACTIONS:	
Resolve primary containment penetration configuration and Appendix J problems; ensure compliance to the design/licensing basis and Appendix J.	Complete; problems identified as part of the Primary Containment walkdown have been resolved by design or testing program changes. Primary Containment manual valves and caps have been marked to be readily identifiable in the plant and are controlled in Procedure 2.0.2.
Review containment penetration configurations to ensure that the safety functions as assumed in the safety analysis can be provided by the as-built penetrations.	Complete; review performed from walkdown findings, and design changes implemented as required.
Accelerate the Design Basis Reconstitution effort with risk significant safety systems given priority.	System prioritization complete. Bids being evaluated for expedited DCDs.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 3

APPARENT VIOLATION:

The Licensee identified approximately 300 examples of components associated with the containment penetrations which were not classified as essential.

The failure to design, fabricate, and erect the containment isolation barriers to quality standards that reflect the importance of the safety function was identified as an apparent violation of 10 CFR 50, Appendix B, Criterion III.

DISCUSSION:

- 46 penetrations where piping was classified IIIN, IVP or indeterminant.
- The 46 penetrations involved 262 socket welds and 35 butt welds (297 total).

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 3

SAFETY SIGNIFICANCE:

- All 262 socket welds passed liquid penetrant examination.
- 5 of the 35 butt welds showed rejectable indications, but were analyzed and determined to have been acceptable.
    - The District did repair or remove each of these welds.
- Results of the NDE verified that the as-found condition of piping was acceptable.

REGULATORY SIGNIFICANCE:

- Adverse condition existed for significant amount of time.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 3

MITIGATING FACTORS:

- Identified by the District as a result of broad corrective actions taken in response to IR 93-17 violation.
- Review of piping classification resulted from expanding the issue to ensure it was bounded.
- Misclassification, not a hardware concern.
- Caused by original design, not introduced through design change process.

CAUSE:

- Focus on system function over containment function resulted in original design error.

CONTAINMENT PENETRATIONS (CONT)
APPARENT VIOLATION 1, EXAMPLE 3

CORRECTIVE ACTION:

- Reviewed each containment penetration to ensure proper classification.
- Reviewing ASME Section XI classification boundaries to identify any similar problems.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 1, EXAMPLE 3

CORRECTIVE ACTIONS	STATUS
CAUSES:	
• FOCUS ON SYSTEM FUNCTION OVER PRIMARY CONTAINMENT FUNCTION FOR AFFECTED PIPING RESULTED IN ORIGINAL DESIGN ERROR OF FAILURE TO TRANSLATE PRIMARY CONTAINMENT DESIGN SPECIFICATIONS INTO THE PIPING SPECIFICATION.	
ACTIONS:	
• Identify welds in penetration-attached piping for which original construction NDE was insufficient.	Complete; welds have been identified, construction NDE performed as required, and design documents updated as applicable in accordance with DC 94-212J.
• Review ASME Section XI classification boundaries to identify other potential pressure boundary classification errors.	In progress; a draft Section XI classification boundary basis has been developed and is in engineering review.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 1

APPARENT VIOLATION:

The Licensee determined that the containment isolation valves in 54 penetrations had not had type c local leak rate tests performed on 68 of the components passing through the penetrations. The systems associated with these valves were classified as nonessential. However, the containment isolation valves were required to function to prevent the release of the post-accident containment atmosphere.

The failure to perform Type C Local Leak Rate Tests was identified as an apparent violation of Technical Specification 4.7.A.2.f.1.

DISCUSSION:

- 64 total penetrations identified involving 82 components where Type B OR C LOCAL Leak Rate Testing had not been performed.
- 57 penetrations involving 75 valves (Type C).
- 7 Penetrations involving 7 seals (Type B).

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 1

SAFETY SIGNIFICANCE:

- Valves
  - 28 of 64 penetrations had caps in addition to a single manual valve.
  - 23 of 64 do not connect directly with containment atmosphere.
  - 5 of 64 penetrations were TIP lines.
  - 1 OF 64 was post accident sampling.
  - As-found LLRTs pass.
- Seals
  - 7 of 64 involve passive seals (flanges and sealant compounds).
- All passed as-found LLRT testing except for IA-65CV (LATER - SEE EXAMPLE 2).

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 1

REGULATORY SIGNIFICANCE:

- Violation of 10 CFR 50 Appendix J, Section III.D.2 and 3.

MITIGATING FACTORS:

- District identified as a result of broad corrective actions taken in response to IR 93-17 Violation.
- Resulted from expanding issue to bound problem.

CAUSE:

- Inadequate monitoring and management of Appendix J Test Program.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 1

CORRECTIVE ACTIONS:

- Resolved design and testing problems.
- Established Appendix J Program owner and developing Appendix J Program Basis Document.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 1

CORRECTIVE ACTIONS	STATUS
CAUSES: • INADEQUATE MONITORING AND MANAGEMENT OF APPENDIX J TEST PROGRAM ACTIONS:	
Resolve Primary Containment penetration configuration and Appendix J problems; ensure compliance to the design/licensing basis and Appendix J.	Complete; problems identified as part of the primary Containment walkdown have been resolved by design or testing program changes. Primary Containment isolation components which were not previously Type B or Type C tested have been tested as required and added to the Appendix J test program.
Establish an Appendix J Program Owner.	In progress; a program owner has been assigned and an Appendix J testing basis document is being prepared.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 2

APPARENT VIOLATION:

The total leakage of the Local Leak Rate Tests performed on components previously not tested exceeded the Technical Specification limit for leakage to ensure containment integrity.

This was identified as an apparent violation of Technical Specification 4.7.A.2.f.1.

DISCUSSION:

- Total leakage, excluding IA-65CV less than Technical Specification limit.
- IA-65 CV could not be pressurized in the as-found condition.
- Bench testing could not reproduce the excessive leakage.
- Used conservative method to determine estimated leakage.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 2

SAFETY SIGNIFICANCE:

- Calculated dose to control room and offsite within limits.
- X-22 penetration containing IA-65CV is closed loop inside containment and outside containment; likelihood of large leak concurrent with core damage accident very low (1E-11).
- Even assuming a major leak from X-22, doses within limits.

REGULATORY SIGNIFICANCE:

- Condition existed for extended time.
- An adequate test program would have discovered and corrected this problem.
- The aggregate consequences could have been significant if total leakage had been higher.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 2

MITIGATING FACTORS:

- District identified as a result of broad corrective actions taken in response to IR 93-17 violation.
- Minimal effects on consequences of an accident while it existed.

CAUSE:

- Inadequate monitoring and management of Appendix J Program.

CORRECTIVE ACTIONS:

- Resolved containment design and testing problems, including IA-65CV.
- Established Appendix J Program owner and developing Appendix J program basis document.

CONTAINMENT PENETRATIONS (CONT)

APPARENT VIOLATION 2, EXAMPLE 2

CORRECTIVE ACTIONS	STATUS
CAUSES: • INADEQUATE MONITORING AND MANAGEMENT OF APPENDIX J TEST PROGRAM.	
ACTIONS: • Resolve Primary Containment penetration configuration and Appendix J problems; ensure compliance to the design/licensing basis and Appendix J.	Complete; problems identified as part of the Primary Containment walkdown have been resolved by design or testing program changes. IA-CV-65CV was replaced and the penetration modified to include two check valves in series. A Type C test was performed on this configuration with satisfactory results.
• Establish an Appendix J program owner.	In progress; a program owner has been selected and an Appendix J Testing Basis Document is in the course of preparation.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 3

APPARENT VIOLATION:

The failure to perform local leak rate testing for several instrument pressure switches was identified as an apparent violation of Technical Specification 4.7.A.2.f.1.

DISCUSSION:

- Involved 15 pressure instruments normally exposed to containment atmosphere that were valved-out during ILRT performance.
- These instruments perform containment isolation, ECCS initiation, and SCRAM functions.

SAFETY SIGNIFICANCE:

- Each instrument successfully LLRT tested.
- All perform their safety function at low containment pressure <2 PSIG and would actuate prior to peak accident pressure.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 3

REGULATORY SIGNIFICANCE:

- Condition existed for extended period.

MITIGATING FACTORS:

- District identified as a result of broad corrective actions taken in response to IR 93-17 violation.
- Resulted from expanding issue to bound the concern.
- The instruments were found to be leak tight and capable of performing their containment boundary and system safety functions.

CAUSE:

- Inadequate monitoring and management of Appendix J Program.

CONTAINMENT PENETRATIONS
APPARENT VIOLATION 2, EXAMPLE 3

CORRECTIVE ACTION:

- Resolved containment design and testing concerns.
- Established Appendix J Program owner and developing Appendix J Program basis document.

CONTAINMENT PENETRATIONS (CONT)

APPARENT VIOLATION 2, EXAMPLE 3

CORRECTIVE ACTIONS	STATUS
CAUSES: INADEQUATE MONITORING AND MANAGEMENT OF APPENDIX J TEST PROGRAM. ACTIONS:	
• Resolve Primary Containment penetration configuration and Appendix J problems; ensure compliance to the design/licensing basis and Appendix J.	Complete; problems identified as part of the Primary Containment walkdown have been resolved by design or testing program changes. Surveillance Procedure 6.3.1.1.2 was implemented to perform local leak rate test on the isolated pressure switches, and requisite testing was completed.
• Establish an Appendix J program owner.	In progress; a program owner has been assigned and an Appendix J testing basis document is being prepared.

CONCLUSION

- The District agrees that the three examples identified as apparent Violation 1 constitute a violation of 10 CFR 50, Appendix B, Criterion III.
- The District agrees that the three examples identified as apparent Violation 2 constitute a violation of the Cooper Nuclear Station Technical Specifications.
- The District has provided specific technical information for each example that addresses the safety significance and mitigating factors that relate to each violation.
- While the District has determined that the specific violations may not have individually affected the health and safety of the public, it is acknowledged that the programmatic issues are significant.
- The District is taking aggressive actions to address the broader issues.

TIE-RAP/UNDERVOLTAGE TESTING

D. BUMAN

CHRONOLOGY OF DG TESTING

May 16, 1994	TIE-WRAP Discovered on MCC-N (CR 94-155). - DG #1 Determined to be Operable (OD 94-52). - TIE-WRAP Removed.
May 17, 1994	All Other Switchgear Inspected.
May 23, 1994	Discovered Load Shed Not Specifically Tested.
May 25, 1994	Discovered SWBP Contacts Not Tested - NOUE Declared
May 26, 1994	Started LSFT Review of Selected Systems
May 28, 1994	Expand Scope of Systems
June 3, 1994	Completed Review of LSFT
June 3, 1994	Started Developing TPCNs, PCNs, and Special Procedures to Test Contacts.
June 10, 1994	Commenced Testing
June 17- June 20, 1994	Discovered Problems With Under-Voltage Trip Devices
June 21- June 26, 1994	Performed Troubleshooting of UVTDs
June 26, 1994	Developed Design Change to Replace UVTDs
July 4, 1994	Completed Modifications for Under-Voltage Tripping
July 17, 1994	Contact Testing Scope Expanded
July 21, 1994	Contact Testing Issues Closed for DG #1 - NOUE Ends

TIE-RAP PROCEDURE/POST-MAINTENANCE TESTING

APPARENT VIOLATION 1

Procedure 7.3.2.1 was inadequate in that it failed to provide instructions for the performance of maintenance and testing that would ensure proper operability of the breaker. The failure to provide appropriate instructions for a safety related activity is an apparent violation.

DISCUSSION:

- The procedure was inadequate.
- The inadequate procedure adversely affected safety system operability.
- As-found testing in this procedure is not used to meet Technical Specification requirements.
- The general maintenance performed in this procedure (i.e., wiping, cleaning, lubrication, etc.) and the subsequent test was not performed to support Technical Specification compliance, therefore, should not be considered preconditioning.
- The failure of post-maintenance testing is acknowledged to directly impact the ability of equipment to perform intended safety function.

TIE-RAP PROCEDURE/POST-MAINTENANCE TESTING
APPARENT VIOLATION 94-16-01

SAFETY SIGNIFICANCE:

- Although the installed tie-rap defeated the associated undervoltage trip Device (MCC-N), the diesel generator still would have performed its function.
- Later calculations demonstrate the diesel generators have sufficient capacity to perform their intended safety function with the failure of all undervoltage devices.

REGULATORY SIGNIFICANCE

- Procedures were inadequate to ensure tie-rap removal and post-maintenance operability.

MITIGATING FACTORS:

- Self identified.
- CNS personnel were on a parallel path with the NRC regarding extent of condition and potential safety implications.

TIE-RAP PROCEDURE/POST-MAINTENANCE TESTING
APPARENT VIOLATION 94-16-01

CAUSE:

- Inadequate maintenance procedure.
- Inadequate post-maintenance testing.

CORRECTIVE ACTIONS:

- Walkdowns.
- Review of relevant procedures.

TIE-RAP PROCEDURE/POST-MAINTENANCE TESTING

CORRECTIVE ACTIONS	STATUS
<u>CAUSE:</u>	
• INADEQUATE MAINTENANCE PROCEDURE.	
• INADEQUATE POST-MAINTENANCE TESTING.	
<u>ACTIONS:</u>	
• Walkdowns conducted to verify that no similar cable tie installations were in place. (See 7/28/94 letter to NRC, Item 5(e)1; 8/8/94 letter to NRC, Attachment 1, Item A.1.).	Complete.
• Review performed of CNS mechanical and electrical maintenance procedures; surveillance procedures in chemistry, operations, I&C; as well as the 14.X series I&C procedures, to identify similar procedural deficiencies.	Complete.
• Performed a random review of I&C procedures to ensure that appropriate checks are performed when leads are lifted and landed.	Complete.

TIE-RAP PROCEDURE/POST-MAINTENANCE TESTING (CONT)

CORRECTIVE ACTIONS	STATUS
ACTIONS (CONT)	
• Maintenance Work Practice 5.0.4 revised to add guidance to ensure that any impairments, changes, or blocking devices installed during performance have been removed prior to procedure completion.	Complete.
• Maintenance supervisors held meetings with personnel to reemphasize need for procedural compliance and the need for immediate correction of problems related to incomplete understanding of procedural requirements.	Complete.

LOAD SHEDDING
APPARENT VIOLATION 94-16-02

Technical Specification 4.9.A.1.a requires that a loss of voltage test be performed to demonstrate load shedding from the emergency busses. The failure to perform a surveillance test, as required by the TS, because of inadequate procedures is an apparent violation.

DISCUSSION:

- The procedure was inadequate.

SAFETY SIGNIFICANCE:

- Testing demonstrated all critical contacts able to perform their safety functions.
- Calculations demonstrated if load shedding of non-essential loads did not occur, the diesel generators would have performed their intended safety function.

REGULATORY SIGNIFICANCE

- Long-term inadequate testing has regulatory significance.

LOAD SHEDDING
APPARENT VIOLATION 94-16-02

MITIGATING FACTORS:

- The service water booster pump load shed is only required during torus cooling and RHR testing, which occurs during a low percentage of time.
- Aggressive actions were taken to identify the extent of the deficiencies.
- Components that may not necessarily have been required were conservatively tested.
- Inadequate surveillance testing for the load shedding is another example of a deficient procedure and should not be a separate violation.

CAUSES:

- Inadequate surveillance procedure.

CORRECTIVE ACTIONS:

- Multi-disciplined review of OER program.
- Identified and tested critical contacts.
- Evaluated and/or tested remaining contacts.

LOAD SHEDDING

CORRECTIVE ACTIONS	STATUS
<u>CAUSE:</u>	
INADEQUATE SURVEILLANCE PROCEDURE	
- DID NOT INCORPORATE OER	
- DID NOT FULLY ENVELOPE DESIGN BASIS	
<u>ACTIONS:</u>	
• Performed multi-disciplined review of the Operating Experience program. (See 8/8/94 letter to NRC, Att. 1, Item A.4.B.1.)	Complete.
• Identified and tested critical contacts with automatic safety functions. (See 7/28/94 letter to NRC, Item 5(d)1; 8/8/94 letter to NRC, Attachment 1, Item A.2.)	Complete.
• Performed evaluation and testing of expanded scope of contacts. (See 8/8/94 letter to NRC, Attachment 1, Item A.2.)	Complete.

LOAD SHEDDING

CORRECTIVE ACTIONS	STATUS
ACTIONS:	
• Multi-discipline team has been reviewing past operating experience reviews to determine adequacy. This review includes both external and internal OER. Any safety significant items to be corrected prior to startup.	To be completed prior to startup.

LOAD SHEDDING

CORRECTIVE ACTIONS	STATUS
ACTIONS:	
• Implement provisions to ensure that changes to procedures are reviewed for impact on design input documents. In the interim, the SRG and System Engineering will be instructed to monitor procedure changes for potential impacts to the design basis.	Interim memo issued. Procedure revision due 11/1/94.
• Revise engineering calculation procedure 3.4.7 to include provisions to ensure that design output documents are revised when affected by changes to calculations.	Interim memo issued 8/24/94. Procedure revision due 11/1/94.

LOAD SHEDDING

CORRECTIVE ACTION(S)	STATUS
ACTIONS (CONT):	
• The District initiated a surveillance testing program review that will consist of an independent in-depth review of CNS procedures to:	The validation of the surveillance procedures of the following systems to be completed prior to startup: ADS CS HPCI LPCI RPS
1) Determine if the level of testing is adequate.	
2) Verify procedures are in place to perform all required testing.	
3) Provide recommendations for program and/or procedure improvement.	Scheduled for completed 4/95.

LOAD SHEDDING

CORRECTIVE ACTIONS	STATUS
ACTIONS (CONT):	
• As-found tested 4160 and 480 volt undervoltage devices in breakers that supply electrical loads directly from emergency busses 1F and 1G. (See 7/28/94 letter to NRC, Item 5(c)1.)	Complete.
• Revised 6.3.4.3 to ensure it adequately demonstrates load shedding and sequential loading of the EDGs	Complete.
• Extensive actions taken to provide adequate assurance that logic system functional testing at CNS is adequate. (See 7/28/94 letter to NRC, Item 5(g); 8/8/94 letter to NRC, Attachment 1, Item A.2.)	Complete.
• Comprehensive evaluation of the Electrical Distribution System performed to ensure that problems were enveloped. (See 8/8/94 letter to NRC, Attachment 1, Item A.4.)	Complete.

DIESEL GENERATOR OPERABILITY
APPARENT VIOLATION 94-16-03

Technical Specification 3.9.A.1.b requires that both EDGs shall be operable. As a result of the EDGs not being capable of supplying sufficient power for vital electrical loads, the EDGs were not operable because they could not perform their intended design function.

DISCUSSION:

- The EDGs would have performed as designed regardless of unreliable UVT devices.
- A failure to properly load shed would not have aggravated the consequences of any other known deficiency.
- Other deficiencies are the mis-adjustment of the 59 relay (IR 93-28) setpoint and the setpoint drift of the 27X3-1G device.

DIESEL GENERATOR OPERABILITY
APPARENT VIOLATION 94-16-03

59/27X3 CUMULATIVE EFFECT

	DG1	DG2
July 93 (End of Outage)	- 59 Relay setpoint Improperly raised.	- 59 Relay setpoint Improperly raised.
	- 27X3-1F measured at 10.48 seconds.	- 27X3-1G measured at 9.84 seconds.
November 93	- 59 Relay setpoint corrected.	- 59 Relay setpoint corrected.
June 94	- 27X3-1F measured at 10.69 seconds.	- 27X3-1G measured at 14.46 seconds.

- Worst case cumulative effect of the 59 relays and the excessive drift of 27X3-1G precludes an analytical determination that the DG2 output breaker would have automatically closed upon an initiation signal.
  - DG1 remained unaffected.
  - DG2 output breaker could have been manually closed in accordance with emergency procedures.

DIESEL GENERATOR OPERABILITY
APPARENT VIOLATION 94-16-03

SAFETY SIGNIFICANCE:

TIE-WRAP /UV DEVICE:

- Both diesels were capable of performing their intended safety function even if all non-essential loads remained on the bus (were not load shed).

27X3/59 RELAY INTERACTION:

- The cumulative effect of the improper high voltage permissive setpoints and the drift of 27X3-1G, along with all other worst case drifts and timing inaccuracies could have prevented the DG2 output breaker from automatically closing during a DBA from June to November 1993.
- EOPs require operators to confirm that DG is operating properly.
- DG2 output breaker could have been manually closed.
- The concurrent worst case drift by all relays is extremely unlikely.
- Voltage permissive relays were properly set in November of 1993.
- Successful sequential load test demonstrated the proper function on July 14, 1993.
- DG1 remained operable.

DIESEL GENERATOR OPERABILITY
APPARENT VIOLATION 94-16-03

REGULATORY SIGNIFICANCE:

- Need for additional analysis.

MITIGATING FACTORS:

Tie-Wrap Undervoltage Issues:

- Self identified.
- Broad and extensive corrective actions already discussed.

27X3/59 Relay Interaction:

- Self identified.

CAUSE:

- Cause has been addressed in that this violation is the result of previous violations already discussed.

CORRECTIVE ACTIONS:

- Relay 27X3-1G reset.
- Increased surveillance frequency.

CONCLUSION

- These issues were self identified.
- The preventative maintenance procedure that installed the tie-rap was deficient.
- The load shed surveillance procedure was deficient.
- The procedure deficiencies should be considered two examples of the same violation.
- The UVT devices failing to trip did not render the diesel generators inoperable.
- The interaction between the improperly set 59 device and the drift of the 27X3-1G relay resulted in DG2 operability being indeterminant for the period from July To November 1993.
- The load shed issues and the more recently discovered relay issue are independent.

CLOSING REMARKS

J. H. MUELLER

CLOSING REMARKS

- Violations reinforce the need for strong corrective actions beyond individual issues.
- Must improve fundamental knowledge and set high standards for staff.
  - Safety Culture Training.
  - Field Coaching Team.
- Focus of management expectations.
  - SORC effectiveness.
  - Strong and involved QA.
  - Instinctively making correct choices.
  - Being self-critical.
- Problems are know, priorities and action plans are in place.
- Positive results to date.

DCS

Nebraska Public Power District

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GUY R. HORN
Vice-President, Nuclear
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NLS950028
January 18, 1995

Director, Office of Enforcement
U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Reply to a Notice of Violation and Proposed Imposition Civil Penalties; NRC Inspection Report Nos. 50-298/94-14, 50-298/94-16, and 50-298/94-19; Cooper Nuclear Station, NRC Docket 50-298, DPR-46

Reference: Letter from Mr. J. L. Milhoan (USNRC) to Mr. G. R. Horn (NPPD), dated December 12, 1994, Notice of Violation and Proposed Imposition of Civil Penalties - $300,000 (NRC Inspection Reports 50-298/94-14, 50-298/94-16, and 50-298/94-19).

This letter, including Attachments 1 and 2, constitute Nebraska Public Power District's (the District) reply to the referenced Notice of Violation (NOV) and Proposed Imposition of Civil Penalties in accordance with 10 CFR 2.201. Attachment 2 is a certified check in the amount of $300,000, for payment of the civil penalties. Per conversation with Mr. G. F. Sanborn, the submittal date of this response was extended to January 18, 1995.

The referenced inspection reports document the results of three NRC inspections conducted from May 23 through August 12, 1994, to specifically review the circumstances regarding: (1) the identification, in May and June, 1994, by the Cooper Nuclear Station (CNS) staff of numerous Primary Containment penetrations that were installed in configurations that were contrary to NRC requirements and had never been local leak rate tested, (2) the extent of the conditions causing both Emergency Diesel Generators (EDGs) to be declared inoperable, which had resulted in the declaration of a Notification of Unusual Event and plant shutdown on May 25, 1994, and (3) the identification by the CNS staff of numerous hardware deficiencies that resulted in the failure of the Control Room envelope pressurization test on April 11, 1994.

As discussed in Attachment 1 to this letter, each of the "problems" and violations had common causes relating to NPG management and culture. As you are aware, NPG management changes have been implemented to address these issues. The breadth of personnel changes reflect the District's commitment to incorporating broad industry experience and achieving higher performance standards. More specifically, management oversight of the Condition Reporting Process has been increased to ensure adequate rigor and urgency are applied in the evaluation of non-conforming plant conditions as they are

IE14

Rec'd w/check 1/24/95

identified. Also, focused attention is being placed in the review of Operational Experience, a program that is considered key in early identification and timely resolution of potential issues.

Management communications have been improved through: (a) training for NPG Managers and Supervisors on important management topics including teamwork and communications, (b) daily management meetings to communicate priorities and standards, and to ensure inter-departmental coordination, and (c) implementation of a revised corrective action program that provides focus on organizational, programmatic and human performance concerns.

In summary, the District has taken, and will continue to take aggressive actions responsive to the management issues that have negatively impacted the NPG safety culture. Steps taken that have resulted in the progress made to date have been significant, and the District believes that the CNS "culture" issue is no longer a factor with regard to the recurrence of these violations. The majority of issues and corrective actions discussed herein have been addressed in the July 28 and August 8, 1994 responses to NRC Confirmatory Action Letters (CAL) dated May 27, June 16, July 1, and August 2, 1994; during the September 16, 1994 Enforcement Conference; and in the November 7, 1994 letter to the NRC. As such, the District has not addressed in this response, any perceived differences of opinion since the cited violations should not have occurred in any event, and since they required significant attention before CNS could be considered ready for restart.

Should you have any questions concerning this matter, please contact my office.

Guy Horn

G. R. Horn
Vice-President - Nuclear

Attachments

cc: Regional Administrator
USNRC Region IV

NRC Resident Inspector
Cooper Nuclear Station

NPG Distribution

STATE OF NEBRASKA)
)
PLATTE)

G. R. Horn, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this response on behalf of Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

G. R. Horn

Subscribed in my presence and sworn to before me this 18TH day of January, 1995.

NOTARY PUBLIC

GENERAL NOTARY-State of Nebraska
RITA L. GOFF
My Comm. Exp. Dec. 19, 1995

REPLY TO DECEMBER 12, 1994 NOTICE OF VIOLATION AND
PROPOSED IMPOSITION OF CIVIL PENALTIES - EA NOS. 94-164, 94-165, 94-166
COOPER NUCLEAR STATION
NRC DOCKET NO. 50-298, LICENSE DPR-46

During NRC inspections conducted from May 23 through August 12, 1994, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedures for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the Nuclear Regulatory Commission proposed civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, 42 U.S.C 2282, and 10 CFR 2.201.

For each of the violations and "problems," inadequate management performance significantly contributed to the existence or perpetuation of the deficiencies. Therefore, management related corrective actions addressed in the cover letter to this response and in other District correspondence to the NRC (i.e., District letters to the NRC dated July 28, 1994 (NLS940001); August 8, 1994 (NLS9400026); and November 7, 1994 (NLS940111)) should be considered part of the District's corrective actions. For brevity, this corrective action is not restated in each violation response.

The particular violations and the District's replies are set forth below:

PROBLEM AREA A- Primary Containment Integrity Violations

I. Violation A.1

Violation A.1 contained in Reference 1 cites the following:

"Technical Specification 3.7.A.2.a, "Containment Integrity," states, in part, that "primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel..."

"Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," states, in part, that "...local leak rate tests (LLRTs) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals no greater than two years.... The total acceptable leakage rate for all valves and penetrations other than the MSIVs [main steam isolation valves] is 0.60 La.

"Technical Specification 1.Y, "Surveillance Frequency," states, in part, that "performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO [limiting condition for operation] unless otherwise required by the specification."

"Contrary to the above, from January 18, 1974, until May 27, 1994, primary containment integrity was not maintained at all times when the reactor was critical or when the reactor water temperature was above 212°F and fuel was in the reactor vessel in that the Surveillance Requirement for the local leak rate testing of 82 components had never been implemented at an interval not to exceed two years. As the result of testing conducted on June 23, 1994, Isolation Valve IA-65CV (one of the 82 components) failed the LLRT, resulting in a total leakage value that significantly exceeded the 0.60 La limit. The 0.60 La limit corresponds to a leakage rate of 5.37 scmh (189.60 scfh). The LLRT failure of Valve IA-65CV resulted in a total leakage rate that exceeded 17.66 scmh (623.57 scfh)."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The immediate cause for not performing local leak rate testing (LLRTs) on 82 components is that they had not been previously identified for inclusion in the CNS 10 CFR 50 Appendix J Program. As discussed in LER 94-011 Revision 1, the root cause for this failure was lack of management commitment to program implementation, in that the organizational focus for problem identification and resolution was primarily compliance-based. This resulted in insufficient attention being paid to evolving regulatory issues in this area, for which a compliance-based standard was inappropriate.

When the CNS Operating License was issued (1974), the specific testable Primary Containment penetrations and Primary Containment Isolation Valves (PCIVs) were listed in the CNS Technical Specifications. Eventually, this list was shifted to the Updated Safety Analysis Report (USAR) through a License Amendment. The original licensing basis was that performance of these specific tests constituted a method for Appendix J compliance that was acceptable to the Atomic Energy Commission (AEC). The mindset was that since the design of many of the CNS systems pre-dated the issuance of Appendix J for public comment, adherence to the Technical Specification/USAR list (along with other specific commitments that might be made) was all that was required, despite later regulatory positions that contradicted this approach. Although these deficiencies were self-identified as a result of broad corrective action in response to Inspection Report 93-17, the District duly acknowledges that this condition should have been recognized and corrected long ago.

Corrective Steps Taken and the Results Achieved

In addition to the generic NPG culture improvements that address the root cause as previously noted, other more programmatic corrective actions have been taken. As stated in LER 94-011, Revision 1, walkdowns of the Primary Containment penetrations have been performed. This activity contributed to the District's confidence that the scope of the Appendix J non-compliance has been comprehensively identified. See also, Confirmatory Action Letter Response dated July 28, 1994.) The following courses of action were then followed:

(1) As-found testing was performed for penetrations that had not previously been Type A, B, or C tested and for which this testing was determined to be immediately practicable. Those that were not tested were either modified and then tested, or were designated as candidates for Appendix J exemption. The total as-found leak rate for the testable penetrations, except X-22 which contained drywell pneumatic supply check valve IA-CV-65CV, was 26 SCFH. With regard to penetration X-22, leakage through IA-CV-65CV was significant, preventing pressurization of the penetration using normal leak rate testing apparatus. Accordingly, worst-case leakage was examined during safety consequence assessments that were performed.

    Penetration modifications and component maintenance/replacement were performed. Subsequent testing has verified that the total Primary Containment as-found leak rate is less than the Technical Specification limit.

(2) Design changes were implemented, which included addition of test connections, installation of welded caps on spare penetrations,

complete redesign of some containment isolation barriers, and installation of caps on vents, drain lines and test connections. These design changes have been completed.

(3) For penetrations and components that have been deemed impractical to test in accordance with the requirements of Appendix J, NRC exemptions have been obtained.

Corrective Steps That Will Be Taken to Avoid Further Violations

Appendix J compliance accountability has been improved by redefining "program owner" responsibilities to require the primary duty to be ensuring the integrity of the overall program, rather than merely functioning as a testing facilitator. To assist in this objective, the current licensing basis for Appendix J testing will be formally captured in an Appendix J testing basis document. This will provide a readily available, controlled source of comprehensive information to the Appendix J program owner which will facilitate the correct disposition of future Appendix J issues as they occur.

Date When Full Compliance Will Be Achieved

CNS is now in full compliance with the testing requirements necessary to demonstrate Primary Containment Integrity.

II. Violation A.2

Violation A.2 contained in Reference 1 cites the following:

"10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents."

"CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11, written to implement the requirements of 10 CFR Part 50, Appendix B, Criterion XI, requires that each type of test program performed will be defined by written procedures and instructions, and it requires that acceptance tests will be developed for structures, systems, and components to demonstrate their capability to perform satisfactorily following repairs or modification.

"Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. Specifically as of May 14, 1994, 68 components passing through 54 primary containment penetrations, each required to be local leak rate tested in accordance with the requirements of Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," had not been identified in a procedure as requiring local leak rate testing, as required by CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11. These components had never been local leak rate tested.

"Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. As of June 21, 1994, instrument pressure switches PC-PS-12A, B, C, and D; PC-PS-101A, B, C, and D; PC-PS-119A, B, C, and D; PC-PS-16; and PC-PT-512A and B, each required to be local leak rate tested in accordance with the requirements of

Technical Specification Surveillance Requirement 4.7.A.2.f.1, "Leak Rate Testing," had not been identified in a procedure as requiring local leak rate testing after being isolated from the containment integrated leak rate test, as required by CNS Quality Assurance Program for Operations Policy Directive, Revision 10, Section 2.11. These switches had never been local leak rate tested."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The causes for not establishing written LLRT procedures for the components listed in the violation are identical to those discussed in response to Violation A.1. A lack of rigor in the Appendix J program (caused by a compliance based management philosophy) resulted in not identifying all of the components for which Type C LLRTs were practicable, and in isolating components which should have been left unisolated for inclusion in the Type A ILRT boundary. Had all components within the Appendix J scope been previously identified and properly evaluated, they likely would have been included within this testing control program. Please refer to the discussion under Violation A.1 for a more complete review of this issue.

Corrective Steps Taken and the Results Achieved

The corrective actions discussed for Violation A.1 describe the efforts that have been made to comprehensively identify the additional components that require Appendix J testing. These efforts have been completed, and changes to the CNS Appendix J testing procedures have been made which reflect current licensing basis testing requirements.

Corrective Steps That Will Be Taken to Avoid Further Violations

As discussed in the District's response to Violation A.1, a testing basis document is being prepared that will provide a clear connection between the Appendix J requirements, the CNS licensing basis with respect to Appendix J compliance, and the testing procedures that implement the CNS licensing basis.

Date When Full Compliance Will Be Achieved

CNS is now in full compliance with the requirement to have written procedures that encompass the full scope of the 10CFR50 Appendix J testing program.

III. Violation A.3

Violation A.3 contained in Reference 1 cites the following:

"10 CFR Part 50, Appendix B, Criterion III, "Design Control," states in part, that "[m]easures shall be established to assure that...the design basis...[is] correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

"Draft General Design Criterion 53, a measure written to comply with the requirements of 10 CFR 50, Appendix B, Criterion III, as committed to in Appendix F of the Updated Safety Analysis Report (USAR), states that "[p]enetrations that require closure for the containment function shall be

protected by redundant valving and associated apparatus.

"Draft General Design Criterion 1, as committed to in Appendix F of the USAR, states that "[t]hose systems and components of reactor facilities which are essential to the prevention of accidents which could effect the public health and safety or mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed."

"General Electric Design Specification No. 22A1153, "Codes and Industrial Standard, "Revision 1, states, in Note 3 of the Appendix, that "[p]iping, which is an integral part of the primary containment for isolation purposes, shall have at least the same quality and levels of assurance as the primary containment."

"Contrary to Criterion III, the licensee did not assure that the above design bases were correctly translated into specifications and instructions and did not assure that deviations from quality standards were controlled. Specifically:

a. As of May 14, 1994, numerous primary containment penetrations had no redundant valving. These penetrations included, but were not limited to, Penetrations X-21, X-22, X-25, X-29E, X-30E/F, X-33E/F, X-209A/B/C/D, and X-218.

b. As of February 22, 1994, 10 penetrations consisting of manually operated vents, drains, and test connections and requiring closure for the containment function, had a single manual isolation valve for containment isolation as opposed to the required redundant valving and associated apparatus.

c. During an NRC inspection conducted June 13, 1994, through August 12, 1994, it was determined that approximately 300 containment penetrations had not been designed, fabricated, or installed to the same standards as the primary containment because these components had not been correctly classified as essential. As a result, these penetrations had not been designed, fabricated, or erected to quality standards that reflect the importance of the safety function to be performed (e.g., some welds were not nondestructively tested, some penetrations were not local leak rate tested, and the penetrations were not treated as safety-related by the licensee's quality assurance program)."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

As noted in this Violation, there were two areas of non-conforming Primary Containment penetration design: (1) the lack of Primary Containment penetration barrier redundancy for all process lines passing through the Primary Containment, and (2) the improper classification and maintenance of many penetrations as non-essential. These areas are discussed in more detail below.

(1) Barrier Redundancy- The cited Primary Containment penetration barrier redundancy discrepancies involve manual valve configurations on penetrations designated for process lines. As noted by the NRC in Enclosure 2 of the NOV, these configurations were part of the original plant design. Specifically, the configurations were built to be in conformance with the 1967 Draft AEC General Design Criteria

(GDC), which did not explicitly require redundancy for process lines isolated by manual valves. In addition, the response to Final Safety Analysis Report (FSAR) Question 5.5 clearly identified the District's position with respect to GDC 55 and 56, and Safety Guide 11. In response to Question 5.5, the District stated that a single manual valve would be employed for instrument lines and lines to control systems or devices inside the Primary Containment, including pneumatic lines for valves, dampers, etc. (Regulatory Guide 1.11 still permits the use of single manual valves for some applications and the GDC still provides for the acceptability of containment isolation provisions "on some other defined basis".)

The CNS SER makes it clear that the AEC's technical review for initial licensing was performed based on 10CFR50 Appendix A criteria, within which, Criterion 55 provides more prescriptive redundancy requirements. It is unclear to what degree the AEC reviewed the redundancy issue, but the conclusion was reached in Section 3.1 of the SER that the plant conformed to the intent of the Appendix A criteria. The failure of the NPG to resolve the ambiguity of the licensing basis was a key factor in the perpetuation of non-redundant Primary Containment penetration barrier configurations, both in the original design, and after various design changes.

In two of the cited examples, it is apparent that there was a lack of rigor in maintaining configuration controls. Therefore, apart from the licensing basis ambiguities, inappropriate configuration management contributed to the existence of non-redundant or unqualified barriers.

(2) Penetration Classification- The penetration misclassifications resulted from an original design error, in that, the requirements of General Electric Specification 22A1153 for piping passing through the Primary Containment were not correctly translated into the piping specifications. The passive function of helping to maintain Primary Containment integrity was not recognized as a safety function for otherwise non-essential piping. As a result, piping segments were inappropriately procured, fabricated and maintained to the requirements of USAS B31.1, rather than to a level of quality commensurate to that applied for the Primary Containment (as in USAS B31.7). The impacts of this discrepancy were that in certain instances: (a) material traceability was not maintained as applicable for the piping and components; (b) appropriate non-destructive evaluations (NDE) were not performed on all applicable welds and piping; and (c) applicable non-destructive testing (NDT) was not performed on pressure retaining welds.

Although both of these non-conforming areas were self-identified as a result of design basis reconstitution efforts and broad corrective actions taken in response to Inspection Report 93-17, the District recognizes that these conditions should have been identified and corrected much earlier. Similar to the previous two violations, the failure to more promptly identify and correct these deficiencies was due in part to a compliance-based focus, such that, undue reliance was placed in the continuing adequacy of the original plant design, based on AEC approval during initial licensing.

Corrective Steps Taken and the Results Achieved

The District addressed actions regarding barrier redundancy and penetration classifications in a letter dated August 8, 1994. The following is a summary of corrective actions noted in that letter, as well

as additional responsive activities.

(1) Barrier Redundancy- Walkdowns of Primary Containment penetrations were conducted which verified the as-built barrier configurations. As a result of identified discrepancies associated with inadequate Primary Containment penetration barrier redundancy, design changes have been developed and completed which have brought them into conformance. Also, programmatic enhancements have been made to control vent, drain, and test line barrier configurations as discussed in the District's response to Inspection Report 94-03.

(2) Penetration Classification- A document review was performed for all Primary Containment piping and instrument penetrations to determine the scope and extent of the misclassifications. Welds in penetration-attached process lines, for which original construction NDE was insufficient, were identified. Those that were found to be in non-compliance or indeterminate were subjected to additional NDE. Five welds were found to have rejectable NDE indications and were repaired or replaced as deemed appropriate. The piping and instrument line segments up to and including the PCIVs have been determined via a Design Change to be of equivalent quality to the Primary Containment, and a reconciliation has been performed between USAS B31.1 and B31.7 for these segments. These penetrations and components are now treated as Essential IIN, and will be maintained under the District's ASME Section XI Program.

Corrective Steps That Will Be Taken to Avoid Further Violations

No further directly related corrective steps are planned. However, as stated during the Enforcement Conference, the two ongoing actions described below help to prevent recurrence of similar types of violations.

(1) To help identify other licensing basis issues stemming from the original design, the design basis reconstitution effort is being expedited.

(2) The ASME Section XI boundaries are being reviewed to identify and resolve other potential pressure boundary classification errors. A Section XI classification boundary basis document is being prepared that identifies the Section XI boundary and defines its bases.

Date When Full Compliance Will Be Achieved

CNS is now in full compliance with the requirements for Primary Containment penetration safety classification and PCIV redundancy, as the District understands that the one remaining single-valve PCIV process line is acceptable to the NRC.

PROBLEM AREA B- Operability of the 480 Volt and 4160 Volt Buses

I. Violation B.1

Violation B.1 contained in Reference 1 cites the following:

"Technical Specification 3.9.A.1.c, "Auxiliary Electrical Equipment," requires, in part, that the reactor shall not be made critical from a Cold Shutdown Condition unless the 4160 volt critical buses 1F and 1G and the 480 volt critical buses 1F and 1G are energized, and the undervoltage and loss of voltage relays, as well as their auxiliary relays, are operable.

"Technical Specification Surveillance Requirement 4.9.A.1.a, "Emergency

Buses Undervoltage relays," states that "once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators." USAR Section 2.2.7.2.1.a, "Standby A-C Power (Diesel Generators) Test Capability," defines the function of the protective scheme as providirg for the clearing the buses of all motor loads excepting supply to the 480 volt critical unit substation.

"Technical Specification 1.Y, "Surveillance Frequency," states, in part, that "performance of a Surveillance Requirement within the specified time interval shall constitute compliance with operability requirements for an LCO [limiting condition for operation] unless otherwise required by the specification.

"Contrary to the above, from January 18, 1974, until May 25, 1994, the reactor had been made critical without 4160 volt critical buses 1F and 1G, and 480 volt critical buses 1F and 1G being operable in that the undervoltage relays associated with several of the electrical loads supplied by these buses had never been tested to demonstrate their operability or upon testing, failed to perform their intended function of shedding their respective electrical loads from these buses."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The root cause of this violation was the CNS failure to view existing programs and methods with a self-critical and questioning attitude. With respect to fulfilling the surveillance requirements that demonstrate operability, this resulted in surveillance procedures that did not fully test the load shedding function. This function encompasses the circuit path from the sensing of undervoltage on the 4160 volt and 480 volt busses through the opening of the respective circuit breakers on undervoltage. The requirement to test this function stemmed from a 1979 License Amendment that specifically incorporated this Surveillance Requirement into the CNS Technical Specifications. The reconciliation performed in 1979 between the load sequence testing procedures and these new requirements was unsatisfactory in that all the required loads were not verified to shed on undervoltage (or loss of voltage), and that the logic system functional testing (LSFT) associated with actuation of the undervoltage relays was not comprehensive.

Numerous opportunities occurred for earlier identification. Most notably during the design basis reconstitution effort, through NRC information to the industry on Westinghouse DB-50 circuit breakers and deficient safety-related LSFTs, and through an Electrical Distribution System Functional Inspection (EDSFI) performed at CNS. However, given the NPG organizational focus that was previously in place, these opportunities were not utilized as vehicles for broader programmatic inquiry.

Corrective Steps Taken and the Results Achieved

Several steps were taken to address this violation:

(1) LSFTs for 4160 volt buses 1F and 1G, and 480 volt buses 1F and 1G were satisfactorily performed.

(2) The applicable surveillance procedures were revised to verify that the load shedding function occurs as required.

(3) An electrical calculation was performed which demonstrated that even if load shedding of all of the non-essential 480 volt loads failed to occur, the diesel generators would have performed their intended safety function.

To correct the long-term reliability issues associated with the load shedding capability of Westinghouse DB-50 circuit breakers, a Design Change was implemented which resulted in replacement of the undervoltage trip devices with shunt trip devices within those safety-related circuit breakers that are credited with shedding a non-essential load. Industry experience has shown this to be an effective configuration.

Corrective Steps That Will Be Taken to Avoid Further Violations

This violation has prompted a broader inquiry into the adequacy of the CNS Surveillance Testing Program. The intent of this effort is to verify that all of the surveillance testing requirements have been correctly translated into the surveillance procedures. This effort is more fully discussed in the corrective action for Violation B.2.

Date When Full Compliance Will Be Achieved

CNS is now in full compliance with the requirements of Technical Specifications 3.9.A.1.c and 4.9.A.1.a.

II Violation B.2

Violation B.2 contained in Reference 1 cites the following:

"10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptable limits contained in applicable design documents."

"Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptable limits. Specifically:

a. During an NRC inspection conducted May 23, 1994, through August 12, 1994, Procedure 6.3.4.3, "Sequential Loading of Emergency Diesel Generators," Revision 31, which is performed to satisfy Technical Specification Surveillance Requirement 4.9.A.1.a, "Loss of Voltage Relays," was determined to be inadequate because it did not assure that the emergency diesel generators and critical buses would perform satisfactorily in service in that the procedure did not contain requirements to verify that the 480-volt supply breakers for safety-related and nonsafety-related loads would shed from their electrical buses within a specified time, nor did the procedure identify that the control rod drive pump motors and station air compressors were required to be shed from the electrical bus.

b. During an inspection conducted May 23, 1994 through August 12, 1994, the NRC identified that Procedure 6.3.20.1, "RHR Service Water Booster Pump Flow Test and Valve Operability Test," Revision 27, did not provide for the testing of the load shedding feature of the supply breakers associated with the 4160 volt residual heat removal service water booster pumps."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The root cause for not establishing written test procedures that adequately reflect the Technical Specification requirements is attributable to the same NPG cultural issues discussed in Violations A.1 and B.1. Furthermore, as discussed in Violation B.1, this generic cause manifested itself through:

(1) Not ensuring the incorporation of all load shedding verifications (including associated LSFTs) into the surveillance procedures when they first became recognized as surveillance requirements.

(2) Failure to recognize and correct the procedural deficiency in a more timely manner, particularly with respect to the opportunities that occurred that might have prompted such recognition. These included the design basis reconstitution effort, industry operational experience with respect to Westinghouse DB-50 circuit breakers and inadequate LSFTs, and a previous EDSFI.

Corrective Steps Taken and the Results Achieved

The surveillance procedures cited in this violation have been revised to reflect appropriate load shed testing. Additionally, as discussed in the District's August 8, 1994, response to an NRC Request for Additional Information, the investigation into the deficiencies of Surveillance Procedure 6.3.20.1 prompted a review of the Logic System Functional Testing performed for several key safety systems. This review has identified significant testing omissions that are being addressed as documented in LER 94-009.

Corrective Steps That Will Be Taken to Avoid Further Violations

CNS is currently verifying that all surveillance requirements contained in the CNS Technical Specifications have been adequately translated into surveillance procedures. In summary, each Technical Specification surveillance line item is being compared with its analogous implementing procedure to determine exactly how the requirement is met, and whether the procedure is satisfactory. This judgment is being made with reference to various source documents such as elementary diagrams, flow diagrams, the USAR, and Design Criteria Documents, as applicable. Upon completion of this project, CNS will have system packages that fully document compliance with the Technical Specification surveillance requirements.

Date When Full Compliance Will Be Achieved

Verification that the key safety system surveillance requirements are adequately described by written procedures will be completed prior to startup. As discussed in the CNS Phase 1 Plan, these key systems include the Automatic Depressurization System, Core Spray System, High Pressure Coolant Injection System, Low Pressure Coolant Injection System, Reactor Protection System, Standby Gas Treatment System, Control Room HVAC System, and Reactor Building HVAC System. Verification that appropriate written procedures encompass all surveillance requirements will be achieved by July 31, 1995.

PROBLEM AREA C- Operability of the Control Room Emergency Filter System

I. Violation C.1

Violation C.1 contained in Reference 1 cites the following:

"Technical Specification 3.12.A.1, "Control Room Emergency Filter System," states, in part, that "...the Control Room Emergency Filter system...shall be operable at all times when containment integrity is required.

"The Order Confirming Licensee Commitments on Post-TMI Related Issues, dated July 10, 1981, confirms NPPD's commitment to complete NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability." Item III.D.3.4 involves the review of facility design requirements against the Standard Review Plan. The NPPD response to Generic Letter 80-90, dated December 30, 1980, submitted the control room habitability evaluation, which stated, in part, "the CNS control room ventilation system is designed to maintain the control room at about 1/4 in. H_2O [0.031 kPa] positive pressure by supplying air at a high enough pressure that even when system losses and the booster exhaust fan pressures are accounted for, the control room pressure is still positive..."

"A Safety Evaluation Report for the Cooper Station from the Accident Evaluation Branch on NUREG-0737, Item No. III.D.3.4, "Control Room Habitability," dated February 24, 1982, states, in part, that "...the design meets the criteria identified in Item III.D.3.4 of NUREG-0737 and is acceptable.

"Contrary to the above, from June 1989 until April 28, 1994, the Control Room Emergency Filter system was not operable at all times when containment integrity was required in that testing failed to demonstrate that a positive pressure could be maintained in the control room during the periodic performance of the control room envelope pressurization test."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The circumstances surrounding the prolonged inoperability of the Control Room Emergency Filter System (CREFS) were provided to the NRC in LER 94-006. As discussed in this LER, the unrecognized inoperability of CREFS was primarily the result of a plant culture that did not approach operability issues with a self-critical and questioning attitude. Also contributing to the deficiency was an incomplete understanding of the original system design criteria, which led to unsubstantiated reliance on the adequacy of the perceived licensing basis. Several opportunities occurred to address identified deficiencies in the both the design and performance of the system. These opportunities were missed because of a design basis that was not well defined, inadequate testing, a compliance-based approach to operability, and a failure to implement adequate corrective action even though the pressurization test results were marginal. As a result of these collective deficiencies, the system should not have been considered operable.

Corrective Steps Taken and the Results Achieved

As discussed in LER 94-006, corrective actions have been taken that have restored CREFS to operability. Specifically, door seals in the Control Room envelope were repaired, penetrations were sealed and the adjacent building ventilation control systems inspected and repaired. Testing was

performed that confirmed positive pressurization between the range of +0.04" to +0.05" wg with respect to atmospheric pressure.

As discussed in the District's July 28, 1994, letter to the NRC regarding CREFS commitments, the following additional corrective actions have been taken prevent recurrence:

(1) The worst case design basis conditions for Control Room dose has been reassessed, and specific CREFS performance criteria with respect to this scenario has been established and documented.

(2) An operability limit of ≥ +0.03" wg with respect to atmosphere for the Control Room envelope has been established, together with an administrative limit of ≥ +0.04" wg (in contrast to the previous ≥ +0.01" wg acceptance criteria.) Surveillance testing to these limits is being conducted monthly. In the event the administrative limit is not met, the testing frequency would be increased to once every two weeks. The design basis for the control room envelope continues to be "positive pressure."

(3) A design change has been implemented that will eliminate the problem of Control Room and Cable Spreading Room pressure balance.

(4) To provide additional margin to its established design basis, CREFS has been modified to increase ventilation flow and pressurization. Currently, the field work has been completed. Final design change closure is awaiting acceptance testing and NRC approval of the District's proposed Technical Specification Amendment for CREFS.

Corrective Steps That Will Be Taken to Avoid Further Violations

Upon NRC approval of the Technical Specification Amendment concerning CREFS, the operability and administrative limits for Control Room pressurization will be increased.

Date When Full Compliance Will Be Achieved

The installation of the CREFS modification has greatly increased the Control Room pressurization capability. However, CREFS flow is now well in excess of the Technical Specification band of 341 CFM ±10%. The system will be returned to operability and full compliance achieved upon NRC approval the higher band proposed by the District as an amendment to the CREFS Technical Specification.

II. Violation C.2

Violation C.2 contained in Reference 1 cites the following:

"10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

"CNS Quality Assurance Program for Operation Policy Directive, Revision 10, Section 2.11, written to implement the requirements of 10 CFR Part 50, Appendix B, Criterion XI, requires that each type of test program performed will be defined by written procedures and instructions, and it requires that acceptance tests will be developed for structures, systems, and components to demonstrate their capability to perform satisfactorily

following repairs or modification.

"Contrary to the above, the licensee did not assure that all testing was identified and performed in accordance with written test procedures which incorporated the requirements and acceptance limits in applicable design documents. Specifically, from June 1989 until June 1994, Surveillance Procedure 6.3.17.18, "Control Room Envelope Pressurization Test," Revision 4, was not sufficiently detailed in that it did not incorporate acceptance limits to assure that the Control Room Emergency Filter System would perform satisfactorily in service and because the procedure did not prohibit the inappropriate manipulation of pressures in the adjoining buildings as a precondition for conduction the test."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

The root cause of this violation was the CNS failure to view existing programs and methods with a self-critical and questioning attitude. With respect to the subject of this violation, a surveillance procedure resulted in inadequate guidance and acceptance criteria for CREFS operability.

In addition to the cultural issues that provided the general climate for this violation to occur, the CNS design and regulatory history of CREFS resulted in an inconsistent understanding of what the exact relationship was between positive Control Room pressurization and CREFS operability. Pressurization was part of the original system licensing basis (albeit vaguely defined), but had not been specifically included as a surveillance requirement in the CNS Technical Specifications. Given the compliance-oriented focus that was prevalent at the time, this ambiguity resulted in a surveillance procedure that required only nominal pressurization. Moreover, testing conditions were ill-defined, in that, the required pressures of areas outside the Control Room envelope during the test were not specific.

Corrective Steps Taken and the Results Achieved

The District previously addressed several corrective actions in the CAL response dated July 28, 1994. Also as discussed in the corrective actions for Violation C.1, the worst case design basis conditions for Control Room dose has been reassessed, and specific CREFS performance criteria with respect to this scenario has been established and documented.

Surveillance Procedure 6.3.17.18 has been revised to define acceptance criteria reflecting the design basis performance requirements, and to specify the testing conditions required for areas bounding the Control Room envelope.

An amendment to the CNS Technical Specifications has been submitted to the NRC to include demonstration of positive Control Room pressurization to the surveillance requirements of CREFS.

Corrective Steps That Will Be Taken to Avoid Further Violations

This violation represents a deficiency in the Surveillance Testing Program. As discussed in Violation B.2, a comprehensive effort has been undertaken to verify that the surveillance requirements of the CNS Technical Specifications (as well as other license requirements that impact operability) have been adequately translated into surveillance

procedures.

Date When Full Compliance Will Be Achieved

As discussed in Violation B.2, verification that the key safety system surveillance requirements are adequately described by written procedures will be completed prior to startup. Verification of full compliance with having written procedures that encompass all surveillance requirements will be completed by July 31, 1995.

Violations Not Assessed A Civil Penalty

Section II.A contained in Reference 1 states the following:

"10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," states, in part, that "[a]ctivities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

"Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7, written, in part, to implement 10 CFR Part 50, Appendix B, Criterion V, requires that safety-related drawings be included on the safety-related drawing list."

A. Violation 1

"Contrary to the above, during an NRC inspection conducted June 13, 1994, through August 12, 1994, it was determined that safety-related Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," Revision N27, was not included on the safety-related drawing list. As a result of this determination, the licensee subsequently identified 13 other drawings containing safety-related components that were not included on the safety-related drawing list."

Admission or Denial to Violation

The District admits that deficiencies existed in the safety-related list and that 14 drawings were identified that did not have appropriate safety-related components identified.

Reasons For The Violation

The "safety-related list" was initially developed in 1985, according to the premise that it would include the drawings of those systems that had recognized safety and plant availability functions. The purpose of this was to ensure that quality-affecting activities would only be performed with reference to final Status 1 drawings (As-built, Certified as constructed, Certified, or Certified final by vendor and signed), as opposed to Archival (Status 2) or Construction (Status 3) drawings. The list was established as an interim measure until more programmatic changes were completed. Accordingly, Procedure 3.8 was revised to define the three Status Categories, and to proceduralize the requirement that the user ensure that safety-related activities involve only Status 1 drawings. After this was accomplished, the safety-related list had no quality function with regard to this deficiency.

In 1986, a drawing verification project was initiated to validate the as-built status of selected Control Room drawings. The scope of this effort was initially confined to drawings contained on the previously identified safety-related list. Between 1986 and 1988, the list was revised numerous times as additional safety-related components were identified, which

likewise affected the drawing verification project scope.

In 1989, a step was added to Procedure 3.8 to provide a formal mechanism for making additions or deletions to the list, which would in turn signal the Configuration Management Group of additional drawings that should be screened for as-built verification.

During the above processes, information was not completely transmitted between lists and drawings.

Corrective Actions Taken and the Results Achieved

As discussed above, the Safety-Related List currently serves only as a scoping document for as-built drawing verification efforts. This is a function better served by adequate project scoping instructions than by establishment in the CNS procedures. Accordingly, reference to this set of drawings has been removed from the Procedure 3.8. The additional drawings that were identified as containing safety-related components are being separately assessed for inclusion in the as-built verification project.

Corrective Steps That Will Be Taken to Avoid Further Violations

There are no further corrective steps being planned to address this issue.

Date When Full Compliance Will Be Achieved

CNS is in full compliance with the requirement that activities affecting quality be appropriately prescribed by procedures, with respect to the activities described by CNS Procedure 3.8.

B. Violation 2

"Contrary to the above, during an NRC inspection conducted May 23, 1994, through August 12, 1994, Maintenance Procedure 7.3.2.1, "DB-25 and DB-50 Circuit Breakers - Setting, Testing, and Maintenance (With Amptectors)," Revision 3, was determined to be inappropriate to the circumstances in that the procedure did not contain a requirement to remove tie-wraps from the subject breakers following preventive maintenance, nor did the procedure provide for comprehensive post-maintenance testing of all circuit breaker functions following the completion of preventive maintenance."

Admission or Denial to Violation

The District admits the violation.

Reasons for Violation

This violation resulted from the discovery on May 16, 1994 that a tie-wrap was installed on the undervoltage trip device of the feeder breaker to MCC-N. Subsequent investigations revealed that the tie-wrap was installed as allowed during the performance of Maintenance Procedure 7.3.2.1. This procedure was found to have no explicit requirements for removing the tie-wrap, or for post-maintenance testing that would identify such discrepant conditions.

As stated in the District's July 28, 1994 response to Confirmatory Action Letter 4-94-06b, the root cause of the event was the failure of management to ensure that requirements for configuration control were adequately implemented into the maintenance procedure. Management's expectations

were not clearly communicated and effected through the procedure review and approval process. As a result, a requirement to remove the tie-wrap was not included at the conclusion of the procedural section.

A contributing factor to the procedural omission was the inappropriate assumption that such restoration steps were within the "skill of the craft," and as such, did not require specific articulation. In this case, it is clear that restoration steps should have been provided.

Corrective Steps Taken and the Results Achieved

As previously discussed in the CAL response dated July 28, 1994, the following steps have been taken to correct the immediate condition:

(1) Plant walkdowns have been performed that have verified that no other tie-wraps or other blocking devices were installed on any of the 480 volt breakers on 480 volt busses 1A, 1B, 1E, 1F, and 1G.

(2) A review was conducted of station procedures covering electrical and mechanical maintenance to determine if similar ambiguities existed with regard to blocking device removal. This resulted in 18 procedure changes. A similar review was performed for procedures controlled by the Operations, I&C, Engineering, and Radiological Departments, which likewise resulted in a number of procedure changes.

(3) A revision has been made to Maintenance Work Practice (MWP) 5.0.4 to add guidance that any impairments, changes or blocking devices installed during the performance of maintenance activities have been removed prior to completion of the procedure.

(4) Maintenance supervision has communicated to their departments the need for procedural compliance and immediate correction of procedural problems and/or incomplete understanding of procedure requirements.

Corrective Steps That Will Be Taken to Avoid Further Violations

The District is committed to broad-based action to achieve excellence in Configuration Management, as discussed in the NPG Performance Improvement Plans. These actions, in addition to the corrective actions described above will prevent further similar violations.

Date When Full Compliance Will Be Achieved

CNS is now in full compliance with the requirement that activities affecting quality be appropriately prescribed by procedures, with respect to installation and removal of temporary blocking devices.

The licensee currently plans to commence startup and power ascension on February 8, 1995. An interim restart organization has been formed, including a dedicated Restart Manager, 24-hour site management coverage, an augmented operating crew on shift and a continuously-manned, dedicated work control center. The power ascension plan calls for hold points at 50% and 90% power, and includes a contingency shutdown from 30% power for corrective maintenace, if necessary. The licensee plans to reach 100% power in approximately 3 weeks. Region IV has initiated 24-hour coverage at the site and will focus on surveillance testing and maintenance, the areas of weakness that led to the extended shutdown. This augmented coverage will continue through March 5, 1995, as currently planned.

Docket No. 50-298

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