U.S. NUCLEAR REGULATORY COMMISSION

Nu Docs

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

NRC FOIA REQUEST NUMBER(S): FOIA —95-262

RESPONSE TYPE: FINAL [ ] XX PARTIAL #8

DATE: DEC 0 6 1995

DOCKET NUMBER(S) (If applicable)

REQUESTER: Scott J. Patterson

PART I.—AGENCY RECORDS RELEASED OR NOT LOCATED *(See checked boxes)*

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Requested records are available through another public distribution program. See Comments section.
	Agency records subject to the request that are identified in Appendix(es) ______ are already available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.
XX	Agency records subject to the request that are identified in Appendix(es) M are being made available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number.
	Agency records subject to the request that are identified in Appendix(es) ______ may be inspected and copied at the NRC Local Public Document Room identified in the Comments section.
	Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, N.W., Washington, DC.
	Agency records subject to the request are enclosed.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
	Fees
	You will be billed by the NRC for fees totaling $ ______.
	You will receive a refund from the NRC in the amount of $ ______.
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______, No. ______

PART II. A—INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

XX	Certain information in the requested records is being withheld from public disclosure pursuant to the exemptions described in and for the reasons stated in Part II, B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 2120 L Street, N.W., Washington, DC in a folder under this FOIA number.

COMMENTS

The NRC is continuing to review records subject to your request. We will notify you upon completion of the review.

SIGNATURE, DIRECTOR, DIVISION OF FREEDOM OF INFORMATION AND PUBLICATIONS SERVICES

[signature]

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST (CONTINUATION)	FOIA NUMBER(S)	DATE
	FOIA — 95-262	DEC 0 6 1995

PART II.B – APPLICABLE EXEMPTIONS

Records subject to the request that are described in the enclosed Appendix(es) N are being withheld in their entirety or in part under the Exemption No.(s) and for the reason(s) given below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.17(a) of NRC regulations.

1. The withheld information is properly classified pursuant to Executive Order. (Exemption 1)

2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (Exemption 2)

3. The withheld information is specifically exempted from public disclosure by statute indicated. (Exemption 3)
   - Sections 141-145 of the Atomic Energy Act, which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).
   - Section 147 of the Atomic Energy Act, which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).

4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated. (Exemption 4)
   - The information is considered to be confidential business (proprietary) information.
   - The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).
   - The information was submitted and received in confidence pursuant to 10 CFR 2.790(d)(2).

XX 5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. (Exemption 5). Applicable Privilege:
   - XX Deliberative Process: Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency.
   - Attorney work-product privilege. (Documents prepared by an attorney in contemplation of litigation.)
   - Attorney-client privilege. (Confidential communications between an attorney and his/her client.)

XX 6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (Exemption 6)

7. The withheld information consists of records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (Exemption 7)
   - Disclosure could reasonably be expected to interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow recipients to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (Exemption 7 (A))
   - Disclosure would constitute an unwarranted invasion of personal privacy. (Exemption 7(C))
   - The information consists of names of individuals and other information the disclosure of which could reasonably be expected to reveal identities of confidential sources. (Exemption 7 (D))

OTHER

PART II. C – DENYING OFFICIALS

Pursuant to 10 CFR 9.25(b) and/or 9.25(c) of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Freedom of Information and Publications Services, Office of Administration, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL EDO	SECRETARY	IG
James Lieberman	Director, Office of Enforcement	Appendix N/1	XX		
L. J. Callan	Regional Administrator Region IV	Appendix N/2	XX		

PART II. D – APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified there. Any such appeal must be made in writing within 30 days of receipt of this response. Appeals must be addressed, as appropriate, to the Executive Director for Operations, to the Secretary of the Commission, or to the Inspector General, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-95-262

APPENDIX M
DOCUMENTS BEING PLACED IN THE PDR

NUMBER	DATE	DESCRIPTION
1.	Undated	Employee/Position listing (1 page)

Re: FOIA-95-262

APPENDIX N
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRIPTION
1.	10/17/94	Memorandum from Callan to Lieberman, concerning recommended enforcement action (1 page) with attachments (374 pages total) The 1 page memorandum and attachments 1, 2, and 3 are being withheld Exemption 5
	Undated	Attachment 1: Draft enforcement action regarding NPPD concerning EAs 94-164, 165, and 166 (7 pages) Exemption 5
	Undated	Attachment 2: Draft enforcement action regarding NPPD concerening EAs 94-164, 165, and 166 (13 pages) Exemption 5
	Undated	Attachment 3: Draft Commission Paper concerning Civil Penalty to NPPD (2 pages) Exemption 5
	09/09/94	Attachment 4 (1): Letter transmitting Inspection Report 50-298/94-19 (6 pages) PDR Accession No. 9409160239
	05/11/94	Attachment 4 (2): Licensee Event Report 50-298/94-006 (5 pages) PDR Accession No. 9405160239
	10/12/93	Attachment 4 (3): Letter transmitting Inspection Report 50-298/93-17 (6 pages) PDR Accession No. 9310200057
	09/12/94	Attachment 4 (4): Cooper Nuclear Station Bachup Data (6 pages)
	Undated	Attachment 4 (5): 28 assorted pages from the Cooper Nuclear Station's Tech Specs, updated SAR, and NUREG-800

APPENDIX N
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRIPTION
	01/02/80	Attachment 4 (6): Letter transmitting Show Cause Order (2 pages) PDR Accession No. 8001170017
	07/10/81	Attachment 4 (7): Letter transmitting Order (2 pages) PDR Accession No. 8107290130
	02/24/82	Attachment 4 (8): Letter concerning NUREG-0737 issues (4 pages) PDR Accession No. 8203160022
	06/09/87	Attachment 4 (9): Letter concerning Control Room Habitability (2 pages) PDR Accession No. 8706120173
	08/26/87	Attachment 4 (10): Letter concerning Control Room Ventilation (9 pages) PDR Accession No. 8709010108
	Undated	Attachment 4 (11): Unsigned copy of Inspection Report 50-298/94-14. The signed version is in the PDR (21 pages) PDR Accession No. 9409070043
	08/27/93	Attachment 4 (12): Unsigned copy of Inspection Report 50-298/93-99. The signed version is in the PDR (2 pages) PDR Accession No. 9309070112
	06/23/93	Attachment 4 (13): Unsigned copy of the SALP Report. The signed version is in the PDR (31 pages) PDR Accession No. 9306290092
	07/05/94	Attachment 4 (14): Licensee Event Report 50-298/94-011 (5 pages) PDR Accession No. 9407130080
	06/22/94	Attachment 4 (15): Internal Cooper Station memorandum regarding Local Leak Rate Discrepancies (16 pages)
	11/18/66	Attachment 4 (16): General Electric Design Specifications (6 pages)

Re: FOIA-95-262

APPENDIX N
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRI'TION
	Undated	Attachment 4 (17): 60 assorted pages fron the Cooper Nuclear Station's Technical Specifications
	02/14/73	Attachment 4 (18): 4 pages of the Cooper Nuclear Station Safety Evaluation
	09/10/75	Attachment 4 (19): 2 pages of a letter from NPPD to NRC regarding Compliance Requirements
	08/05/75	Attachment 4 (20): 2 pages of a letter from NRC to NPPD regarding Compliance Requirements
	02/14/73	Attachment 4 (21): 4 pages of a Federal Register Notice
	02/17/77	Attachment 4 (22): Letter discussing compliance with 10 CFR Part 50, App J (2 pages) PDR Accession No. 8708180143
	04/04/77	Attachment 4 (23): Letter discussing compliance with 10 CFR Part 50, App J (2 pages) PDR Accession No. 8708180140
	09/16/77	Attachment 4 (24): Letter discussing compliance with 10 CFR Part 50, App J (3 pages) PDR Accession No. 8707160810
	09/16/77	Attachment 4 (25): Safety Evaluation concerning exceptions to 10 CFR Part 50, App J 12 pages) PDR Accession No. 8707160815
	Undated	Attachment 4 (26): 2 pages from the Cooper Nuclear Station's Tech Specs
	10/30/78	Attachment 4 (27): Letter discussing compliance with 10 CFR Part 50, App J (11 pages) PDR Accession No. 7811020253

Re: FOIA-95-262

APPENDIX N
DOCUMENTS BEING RELEASED IN PART

NUMBER	DATE	DESCRIPTION
	Undated	Attachment 4 (28): 23 assorted pages from the Cooper Nuclear Station's Containment Penetration Checklist
	Undated	Attachment 4 (29): 26 assorted pages from the Cooper Nuclear Station's Containment Local Leak Rate Tests
	Undated	Attachment 4 (30): 21 assorted pages from the Cooper Nuclear Station's Updated Safety Analysis Report
	09/09/94	Attachment 4 (31): Letter transmitting Inspection Report 50-298/94-16 15 pages) PDR Accession No. 9409150149
	07/05/94	Attachment 4 (32): Licensee Event Report 50-298/94-009 (5 pages) PDR Accession No. 9406290208
	Undated	Attachment 4 (33): 6 assorted pages from the Cooper Nuclear Station's Tech Specs and Updated Safety Analysis Report
2.	11/10/94	E-mail message from Harrell regarding positions and addresses of Cooper Nuclear Station Employees (1 page) Exemption 6

MORGAN, LEWIS & BOCKIUS

PHILADELPHIA
NEW YORK
MIAMI
PRINCETON
BRUSSELS

COUNSELORS AT LAW
1800 M STREET, N.W.
WASHINGTON, D.C. 20036
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WASHINGTON
LOS ANGELES
HARRISBURG
LONDON
FRANKFURT
TOKYO

June 9, 1995

FREEDOM OF INFORMATION
ACT REQUEST
FOIA-95-262
Rec'd 6-12-95

Carlton Kammerer
Director, Division of Freedom of
Information & Publications Services
U.S. Nuclear Regulatory Commission
Office of Administration
Mail Stop T6E4
Washington, D.C. 20555

Re: Freedom of Information Act Request

Dear Mr. Kammerer:

Pursuant to 5 U.S.C. § 552(a)(3) and 10 CFR § 9.23(b), I request copies of any NRC documents to include, but not limited to, notes, meeting minutes, transcripts, recordings, summaries, electronic messages (E-mail), drafts, reports, and memoranda that contain factual information that formed the basis of, or relate to, the following reports or other documents regarding Nebraska Public Power District's ("NPPD") Cooper Nuclear Power Station (Docket No. 50-298):

(1) NRC Systematic Assessment of Licensee Performance ("SALP") report issued to NPPD in June 1993 for the period January 19, 1992 to April 24, 1993;

(2) Notice of Violation and Proposed Imposition of Civil Penalties, dated March 30, 1993, regarding licensee letter of December 1, 1992 to NRC that was inaccurate and incomplete in material respects;

(3) Notice of Violation and Proposed Imposition of Civil Penalties, and Inspection Report No. 50-298/93-17, dated October 12, 1993;

(4) NRC Operational Safety Team Inspection ("OSTI") Report No. 50-298/93-202, dated December 28, 1993;

(5) Letter from NRC to NPPD, dated January 25, 1994, regarding declining trend in Cooper's performance;

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(6) Confirmatory Action Letter ("CAL") issued to NPPD on May 27, 1994;

(7) Confirmatory Action Letter issued to NPPD on June 16, 1994;

(8) Letter from the NRC to NPPD, dated June 21, 1994, regarding declining trend in Cooper's performance;

(9) Confirmatory Action Letter issued to NPPD on July 1, 1994;

(10) Letter from the NRC to NPPD, dated July 29, 1994, formalizing plans to conduct a special evaluation of Cooper Nuclear Station;

(11) Confirmatory Action Letter issued to NPPD on August 3, 1994;

(12) NRC Inspection Report 50-298/94-14, dated September 2, 1994;

(13) NRC Inspection Report 50-298/94-19, dated September 9, 1994;

(14) NRC Inspection Report 50-298/94-16, dated September 12, 1994;

(15) NRC Inspection Report 50-298/94-18, dated September 14, 1994;

(16) Letter from the NRC to NPPD, dated November 29, 1994, and the Special Evaluation Team ("SET") report regarding interviews and inspections performed from May to September 1994;

(17) Notice of Violation and Proposed Imposition of Civil Penalties, dated December 12, 1994; and

(18) NRC letter to NPPD regarding declining trend in Cooper's performance, dated February 1, 1995.

In addition, please provide any documents that contain factual information that formed the basis of, or relate to, statements made to NPPD officials during discussions between NRC officials and NPPD on the following dates (a brief description of the discussions and persons believed to be involved are indicated in parentheses):

(1) February 18, 1994 (Ronald W. Watkins, NPPD President and CEO, and Leonard J. Callan, NRC Region IV Administrator);

(2) June 23, 1994 (Guy R. Horn, NPPD Vice President nuclear, and Leonard J. Callan, NRC Region IV Administrator);

(3) Week ending July 29, 1994 (NRC Headquarters meeting with NPPD attended by Region IV representatives, including Leonard J. Callan, NRC Region IV Administrator);

(4) September 1, 1994 (Public meeting of NPPD Board of Directors attended by the Cooper Special Evaluation Team Manager);

(5) November 8, 1994 (NRC public meeting held at the Cooper Nuclear Station to discuss NPPD's work to resolve issues necessary for restart of the Cooper plant);

(6) November 17, 1994 (NRC public exit meeting at which SET results were presented by Ellis W. Merschoff, NRC SET Manager, and the Executive Director For Operations, James M. Taylor);

(7) Any enforcement conferences held between NPPD and the NRC between January 1, 1992 and February 21, 1995; and

(8) All NRC restart panel meetings regarding Cooper.

Finally, I request any NRC documents that contain factual information relating to the following NPPD documents, or any versions of these documents:

(1) The Cooper Nuclear Station Near Term Integrated Enhancement Program, dated May 20, 1994.

(2) NPPD draft Business Plan dated May 21, 1994;

(3) NPPD's internal Diagnostic Self-Assessment Team report dated September 1, 1994;

(4) The Cooper Nuclear Station Startup Plan, Revision 1, dated September 15, 1994; and

(5) Cooper Nuclear Station Performance Improvement Plan, Phase 1, Revision 2, dated October 6, 1994;

(6) Cooper Nuclear Station Performance Improvement Plan, Phase 2/3, Revisicn 1, dated December 9, 1994; and

(7) Cooper Nuclear Station Startup and Power Ascension Plan, Revision 3, dated January 31, 1995.

To the extent that the requested information is included in documents or records that contain the advice, opinions or recommendations of NRC staff, please produce all factual information that can be reasonably segregated, in accordance with 10 CFR § 9.19(b).

I agree in advance to pay any fees associated with this request up to $ 500.00. I request that you notify me if the costs will be more than $ 500.00. I can be reached by telephone at (202) 467-7541. Thank you in advance for your assistance.

Sincerely,

Scott J. Patterson
Legal Assistant

sjp

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

14-4(11)

Docket: 50-298
License: DPR-46
EA 94-165

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

SUBJECT: NRC INSPECTION REPORT 50-298/94-14

This refers to the inspection conducted by Ms. P. A. Goldberg and Mr. C. J. Paulk, of this office, and Mr. S. Cha, an NRC consultant, on June 13 through August 12, 1994. The inspection included a review of activities authorized for your Cooper Nuclear Station facility. At the conclusion of the inspection, the findings were discussed with you and those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements.

Based on the results of this inspection, two apparent violations were identified and are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), 10 CFR Part 2, Appendix C. Accordingly, no Notice of Violation is presently being issued for these inspection findings. Please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

The apparent violations are of concern because it is apparent that the primary containment was inoperable for an undetermined period of time. Additionally, it is apparent that there was a breakdown in your design control program, dating back to initial construction, which you have had numerous opportunities to identify and correct. The apparent breakdown in design control contributed to the problems associated with the primary containment, as well as other recent problems identified at the Cooper Nuclear Station.

An enforcement conference to discuss these apparent violations has been scheduled for September 16, 1994. The decision to hold an enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. The purposes of this

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conference are to discuss the apparent violations, their causes and safety significance: to provide you the opportunity to point out any errors in our inspection report: and to provide an opportunity for you to present your proposed corrective actions. In addition, this is an opportunity for you to provide any information concerning your perspectives on (1) the severity of the violation(s), (2) the application of the factors that the NRC considers when it determines the amount of a civil penalty that may be assessed in accordance with Section VI.B.2 of the Enforcement Policy, and (3) any other application of the Enforcement Policy to this case, including the exercise of discretion in accordance with Section VII. You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding these apparent violations is required at this time.

This enforcement conference, which will also address issues involving the control room filtration system (EA 94-164) and the electrical distribution system (EA 94-166), will be open to public observation in accordance with the Commission's continuing trial program as discussed in the enclosed Federal Register Notices (Enclosure 2). Although not required, we encourage you to provide your comments on how you believe holding this conference open to public observation affected your presentation and your communications with the NRC.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

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

Sincerely,

Thomas P. Gwynn, Director
Division of Reactor Safety

Enclosures:
1. Appendix - NRC Inspection Report 50-298/94-14
2. Federal Register Notices

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

Nebraska Public Power District
ATTN: Mr. John H. Mueller. Site Manager
P.O. Box 499
Columbus. Nebraska 68602-0499

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

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

Kansas Radiation Control Program Director

E-Mail report to D. Sullivan (DJS)

bcc to DMB (IE01)

bcc distrib. by RIV:

L. J. Callan
Branch Chief (DRP/C)
MIS System
Branch Chief (DRP/TSS)
RIV File
Senior Resident Inspector - Fort Calhoun
G. F. Sanborn, EO
W. L. Brown, RC
T. F. Westerman
C. Paulk

Resident Inspector
Leah Tremper, OC/LFDCB, MS: MNBB 4503
DRSS-FIPB
Project Engineer (DRP/C)
Senior Resident Inspector - River Bend
F. R. Huey, WCFO EC
J. Lieberman, OE, MS: 7-H-5
P. Goldberg
A. Howell

RIV:RI*	RIV:RI*	C:EB*	D:DRS*	EO*	D:DRP*	D:DRS
PAGoldberg	CJPaulk	TFWesterman	TPGwynn	GFSanborn	ABBeach	TPGwynn
08/08/94	08/05/94	08/18/94	08/19/94	08/23/94	09/01/94	/ /94

*Previously concurred

APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-298/94-14

EA No.: 94-165

License: DPR-46

Licensee: Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska

Facility Name: Cooper Nuclear Station

Inspection At: Brownville, Nebraska

Inspection Conducted: June 13 through August 12, 1994

Inspectors: P. A. Goldberg, Reactor Inspector, Engineering Branch
Division of Reactor Safety

C. J. Paulk, Reactor Inspector, Engineering Branch
Division of Reactor Safety

Accompanied By: G. Cha, Consultant

Approved: ______________________________ ____________
T. F. Westerman, Chief, Engineering Branch Date
Division of Reactor Safety

Inspection Summary

Areas Inspected: Reactive, announced inspection of the licensee's actions concerning containment penetration problems found as the result of reviews and inspections performed by the licensee. In addition, issues related to motor-operated valves and switch calibration for drywell instrumentation were reviewed.

Results:

- As a result of corrective actions for a previously identified violation, the licensee was reviewing the design function of all piping and equipment pressure parts to determine if they were properly classified. This effort was scheduled to be completed in October 1994 and will be evaluated during followup of Enforcement Action 93-137 (Section 2.1).

- The licensee prepared 15 design change packages to bring the containment penetrations into compliance with the draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the Updated Safety Analysis Report, and 10 CFR Part 50, Appendix J. Seven of these design change packages were reviewed and no concerns were identified (Section 2.1.1).

- During the inspection, the inspectors found that Flow Diagram No. 2028, which depicted 80 safety-related components, was not accurate since it failed to include some safety-related components. The failure to include containment isolation valves on the drawing and the failure to identify the drawing as safety-related was identified as an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.1.2).

- 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 (Section 2.1.3).

- 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. (Section 2.1.3).

- 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 Part 50, Appendix B, Criterion III (Section 2.1.4).

- The licensee identified approximately 300 examples of components associated with containment penetrations which were not classified as essential. The failure to design, fabricate, and erect the containment isolation barriers to quality standards that reflected the importance of the safety function was identified as an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.1.5).

- The licensee determined that Containment Isolation Valve RHR-MOV-M027B was not capable of passing its local leak rate test. The licensee decided to move the primary containment isolation function from the leaking valve to another valve. This change was accomplished by use of a safety evaluation that was performed in accordance with 10 CFR 50.59. It was concluded that the licensee's change of primary containment isolation boundary was adequately justified, and appropriate procedural controls were identified (Section 2.2).

- During a review of the licensee's actions concerning the lack of cleanliness inside motor-operated valve limit switch compartments, it was found that the licensee had not entered the recommended corrective actions into the corrective action tracking system. This was a concern because of the lengthy amount of time allowed to pass before the corrective actions were due which increased the chances for similar events to occur (Section 2.2).

- 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 (Section 2.3).

- Unresolved Item 298/9403-01, concerning ten valves used as single manual valves for containment isolation, was closed. These ten single isolation valves without a second barrier were identified as another example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.4).

Summary of Inspection Findings:

- Example 1 of apparent Violation 298/9414-01 was identified (Section 2.1.2).
- Example 2 of apparent Violation 298/9414-01 was identified (Sections 2.1.4 and 2.4).
- Example 3 of apparent Violation 298/9414-01 was identified (Section 2.1.5).
- Example 1 of apparent Violation 298/9414-02 was identified (Section 2.1.3).
- Example 2 of apparent Violation 298/9414-02 was identified (Section 2.1.3).
- Example 3 of apparent Violation 298/9414-02 was identified (Section 2.3).
- Inspection Followup Item 298/9414-03 was opened (Section 2.2.2).
- Unresolved Item 298/9403-01 was closed (Section 2.4).

Attachments:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

During this inspection period. Cooper Nuclear Station was shutdown.

2 ENGINEERING (37550 and 92903)

This inspection was conducted to review Cooper Nuclear Station's actions concerning problems found with containment penetrations. In addition. licensee's actions concerning dirty torque switches on motor-operated valves and time-delay relays for the emergency diesel generators were reviewed.

The inspectors reviewed the licensing basis for the Cooper Nuclear Station in order to evaluate the problems associated with the containment penetrations against the appropriate criteria. The inspectors found that the licensee was committed to the draft "General Design Criteria for Nuclear Power Plant Construction Permits." issued in July 1967. This commitment was documented in the Updated Safety Analysis Report (USAR). Appendix F. The licensee was evaluated and licensed to the draft General Design Criteria. July 1971. and 10 CFR Part 50. Appendix J. as stated in Sections 3.1 and 6.2.3. respectively. of "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Nebraska Public Power District. Cooper Nuclear Station. Nemaha County. Nebraska. Docket No. 50-298." dated February 14. 1973. The inspectors also found that the licensee acknowledged the applicability of the draft General Design Criteria in the draft design criteria document prepared for the containment systems.

With regard to the applicability of 10 CFR Part 50. Appendix B. to Cooper Nuclear Station. 10 CFR 50.54(a)(1) requires. that each plant licensed subject to the quality assurance criteria in Appendix B shall implement pursuant to 10 CFR 50.34(b)(6)(ii) the quality assurance program described or referenced in the safety analysis report. The final 10 CFR Part 50. Appendix B. rule was issued on June 27. 1970. and the operating license for Cooper Nuclear Station was issued on January 18. 1974.

On the basis of the above. the inspectors reviewed the containment penetration issues against the draft General Design Criteria. July 1967. as described in the USAR. Appendix F: 10 CFR Part 50. Appendix B: 10 CFR Part 50. Appendix J: and. applicable licensee procedures. design specifications. and Technical Specifications.

2.1 Containment Penetrations

The licensee prepared Special Procedure 94-202. dated May 17. 1994. "Containment Walkdown." to inspect each primary containment penetration and the piping to the outboard containment isolation barrier. The purpose of the inspection was to support development of the containment design criteria document; to comply with a commitment, made in response to a violation in NRC Inspection Report 50-298/93-17, to review all containment penetrations; and, to support the upgrade of the licensee's program for primary reactor containment leakage testing in accordance with 10 CFR Part 50, Appendix J.

The inspectors reviewed Special Procedure 94-202 and found that licensee inspection of each primary containment penetration, and components which were in the containment isolation system, was required. This inspection was also in support of the preparation of as-built drawings. The inspectors concluded that the procedure was adequate.

During the inspections, the licensee determined that 46, of the 255 primary containment penetrations inspected, had been incorrectly classified nonessential at the time of plant construction and were not contained in the inservice inspection program. In addition:

- The licensee determined that a number of penetrations had not had local leak rate tests performed in accordance with the requirements of 10 CFR Part 50, Appendix J.
- A number of penetrations did not have two containment barriers outside of the primary containment in accordance with draft General Design Criteria, Criterion 53, July 1971.
- A number of instrument lines and valves within the containment pressure boundary were classified as nonessential.
- 294 welds in the containment isolation barriers were found to either never have had nondestructive examinations performed or the qualification records could not be located.

Many penetrations were improperly classified during the construction of the plant. The inspectors attempted to determine how such a problem occurred. While no definite answer was provided, the licensee stated that the architect engineer apparently had missed a note in the General Electric design specification which resulted in the improper classification of containment penetrations and associated components.

The inspectors found that equipment and components classified as essential were designed, fabricated, installed, and tested in accordance with USAS B31.7-1969, "Nuclear Power Piping." Equipment and components classified as nonessential were designed, fabricated, installed, and tested in accordance with USAS B31.1.0-1967, "Power Piping."

On the basis of these codes, the architect engineer designed the equipment and components. The architect engineer, however, apparently missed a note in General Electric Design Specification 22A1153, "Codes and Industrial

Standard," Revision-1. Note 3 of five, to this specification, stated that "[p]iping, which [was] 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."

In Appendix A of the Updated Safety Analysis Report, the licensee provided definitions for the classification of piping and equipment pressure parts. Class C was assigned for "[p]iping and equipment pressure parts . . . for a high integrity system," such as the containment vessel. To meet this classification, the licensee applied the requirements of USAS B31.7-1969 for Class II piping. Therefore, the penetration piping and equipment pressure parts should have been designed, fabricated, installed, and inspected accordingly.

As a result of corrective actions for a previously identified violation, the licensee was reviewing the design function of all piping and equipment pressure parts to determine if they were properly classified. This effort was scheduled to be completed in October 1994 and will be evaluated during followup of Escalated Action 93-137 for violations cited in NRC Report 50-298/93-17.

The inspectors observed 17 liquid penetrant tests of welds that were originally designed, fabricated, installed, and tested in accordance with USAS B31.1.0-1967 rather than USAS B31.7-1969. The inspectors observed one weld that exhibited indication of weld slag. The licensee rejected that weld. Subsequently, the licensee chipped the weld slag off and retested the weld satisfactorily.

The licensee completed the liquid penetrant testing on 260 welds that had been improperly classified without identifying any other weld that was questionable. The inspectors concluded that the licensee had performed testing in accordance with USAS B31.7-1969 for the welds that had no documentation of such inspection during the construction of the plant.

2.1.1 Design Modifications

To address the concerns identified by the licensee's inspections of primary containment penetrations, design changes were prepared. The inspectors reviewed the 7 design change packages, discussed in the following sections, out of a total of 15 which the licensee was preparing to bring the penetrations into compliance with the draft General Design Criteria, Criterion 53, as stated in the USAR, Appendix F, and 10 CFR Part 50, Appendix J. During the licensee's verification and validation of the draft design criteria document for the primary containment, the identification of problems led to a complete scrutiny of all penetrations (approximately 300). As a result of the licensee's efforts, 99 penetrations were identified with problems other than classification. The problems were categorized into 11 types, which ranged from missing caps to inadequate design.

2.1.1.1 Design Change 94-212 Torus Penetration X-218 Modification

Penetration X-218, as-found, consisted of a ball valve on the torus shell with eight thermocouples routed through it. A sealant of unknown composition filled the void and acted as a containment barrier. The thermocouples were installed under Design Change 76-17, Revision 2, but were never placed in service. The design change was later voided because there were no provisions to calibrate the temperature elements and the equipment was abandoned in place. The penetration was not local leak rate testable, and was not on the local leak rate test list.

The design change consisted of removing all thermocouple hardware and the ball valve, and installing a 5.08 cm (2 in) socket welded cap, which would function as a primary containment boundary, hence the penetration would be restored to its original design. The design change was classified as essential and Seismic Class IS. The 5.08 cm (2 in) socket welded cap was purchased as essential material.

The applicable design code for fabrication and installation was USAS B31.7-1969. Weld integrity was checked by 100 percent liquid penetrant nondestructive examination and pneumatically tested to 1.25 of design pressure. The results of the liquid penetrant tests were discussed in Section 2.1 of this report.

The inspectors did not identify any concerns with this design change.

2.1.1.2 Design Change 94-212A Electrical Penetrations X-209A through D Modifications

Design Change 94-212A consisted of two parts: the first part, associated with Penetrations X-209A and X-209C, involved modifying the two thermocouple penetrations to permit periodic local leak rate testing as required by 10 CFR Part 50, Appendix J; and, the second part, associated with Penetrations X-209B and X209D, involved permanently capping the two penetrations.

The inspectors did not identify any concerns with this modification. The inspectors reviewed Design Change 94-212A in its entirety, verified the design changes during the walkdown, and concluded that it was acceptable.

2.1.1.3 Design Change 94-212B Penetrations X-43 and X-44 Testable Flanges

This design change replaced two flanged piping joints near Penetrations X-43 and X-44 with flanges incorporating a testable, double o-ring design. The new design permitted these joints, which were part of the primary containment boundary, to be periodically tested in accordance with 10 CFR Part 50, Appendix J.

The design change was classified essential and Seismic Class IS. All pressure retaining material was procured essential. The inspectors did not identify any concerns with this modification.

2.1.1.4 Design Change 94-212D Penetration X-21 and X-22 Upgrade

The purpose of Design Change 94-212D was to enhance the isolation capacity for both the service air and instrument air headers, upstream of Penetrations X-21 and X-22, respectively. Additionally, the modification provided test connections for periodically performing local leak rate tests of the containment isolation valves in accordance with 10 CFR Part 50, Appendix J requirements.

The inspectors did not identify any concerns with this modification.

2.1.1.5 Design Change 94-212E Primary Containment Integrity Issues

Design Change 94-212E consisted of three parts. The first part removed Swagelok caps and installed valves and caps at ten test connections for instruments which were in direct communication with primary containment. The ten test connections were PC-PT-1B2, -4B2, -5B2, -1A1, -4A1, -5A1, -2104A, -2104B; and PC-PI-2104AG, -2104BG. Also, at PC-DPT-3A1, Drain Valve PC-V-243 was missing and was reinstalled. The second part of the modification removed unnecessary tees located in instrument lines which communicated directly with primary containment and replaced them with unions, elbows or installed welded caps. The third part of the modification cut and capped 14 instrument lines which penetrated primary containment and had previously been spared out. The valves were removed and welded caps installed on the lines at the penetrations.

The inspectors did not identify any concerns with this modification.

2.1.1.6 Design Change 94-212H Post Accident Sampling System Modifications and Penetration X-51F Upgrade

The purpose of Design Change 94-212H was to replace the existing nonessential post-accident sampling system Containment Atmosphere Sample Isolation Valve PAS-AOV-3AV with two qualified 1.27 cm (0.5 in) air-operated valves, PC-AOV-247AV and PC-AOV-248AV, at Penetration X-51F. In addition, test connections with capped manual valves were provided.

The inspectors did not identify any concerns with this modification.

2.1.1.7 Maintenance Work Requests 94-2978 and 94-3116

These maintenance work requests installed caps and plugs to provide the second barrier for containment isolation. During the licensee's inspections, numerous vents, drains, and test connections, having direct access to the primary containment, were found to lack a second barrier. These were identified and a cap or plug was added, depending on the installation.

The inspectors reviewed Maintenance Work Requests 94-2978 and 94-3116 and concluded that both were acceptable.

2.1.2 Drawing Control

During a review of the penetration walkdown packages, the inspectors noted that some of the containment isolation valves identified on the penetration drawings, and existing in the plant, were not included on Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," Revision N27. The inspectors found that 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. The inspectors concluded that the drawing was inaccurately classified as a result of the problems associated with classification of components as discussed in Section 2.1, above.

Cooper Nuclear Station Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7, defined a safety-related drawing as "a drawing or schematic describing the features, characteristics, design or location of safety-related components, systems, or structures." The procedure also stated that any new drawing, or portion of a new drawing, classified as safety-related would be added to the safety-related drawing list.

During the inspection, the licensee initiated Condition Report 94-0309 in response to the inspectors' finding. The condition report stated that the subject drawing depicted a total of 80 safety-related components, but was not included on the safety-related drawing list. In response to this condition report, the licensee identified an additional 13 drawings, with safety-related components, that were not included on the safety-related drawing list.

Additionally, Draft General Design Criteria, Criterion 1, July 1967, in accordance with Appendix F to the USAR, stated that ". . . those systems and components of the station which [had] a vital role in the prevention or mitigation of consequences of accidents affecting the public health and safety [were] designed and constructed to high quality standards . . ."

The inspectors identified five missing valves on Flow Diagram 2028. These valves were associated with Penetrations X-18, X-30E, X-30F, X-33E, and X-33F. For Penetration X-18, an unlabeled vent isolation valve downstream of Valve RW-254 was not on the drawing. For Penetration X-30E, Valve NBI-502, the manual containment isolation valve for the air-to-vessel flange leak off detection air-operated valve, was not shown. For Penetration X-30F, Valve MS-900, the manual containment isolation valve for the air-to-reactor vessel head vent was not shown. For Penetration X-33E, Valve MS-501, the manual containment isolation valve for the air-to-vessel flange leak off detection air-operated valve, was not shown. For Penetration X-33F, Valve MS-899, the manual containment isolation valve for the air-to-vessel head vent, was not shown.

Appendix B to 10 CFR Part 50, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into . . . drawings . . . 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."

The inspectors identified the licensee's failure to properly classify drawings as safety-related and the failure to include safety-related components on the drawing as Example 1 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.3 Local Leak Rate Tests

The licensee determined that the containment isolation valves in 54 penetrations did not have 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 since they did not have to function post-accident.

Containment isolation valves, however, were required to function to prevent the release of the post-accident containment atmosphere. Containment isolation valves, as defined in 10 CFR Part 50, Appendix J, would be "any valve which [was] relied upon to perform a containment isolation function." Type C tests were required for containment isolation valves that "provide[d] a direct connection between the inside and outside of the primary containment under normal operation . . . ; [were] required to close automatically upon receipt of a containment isolation signal . . . ; and [were] required to operate intermittently under post-accident conditions."

In accordance with draft General Design Criteria, Criterion 57, July 1967, as stated in Appendix F to the USAR, the licensee was required to demonstrate the ". . . functional performance of containment system isolation valves and monitoring valve leakage."

Technical Specification 4.7.A.2.f.1 required that "local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves . . ."

The inspectors identified the failure to perform local leak rate tests as Example 1 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

The licensee had begun performing local leak rate tests on the identified components. The inspectors attempted to review the results of this testing. The licensee had not developed a running total of the results of the as-found tests to determine the status of the primary containment and its ability to perform as designed. The inspectors were informed that one penetration (X-22), on June 23, 1994, had in excess of 17 scmh (600 scfh) leakage. This significantly exceeded Technical Specification 4.7.A.2.f.1 leakage limit of 0.60 La (5.37 scmh (189.6 scfh)).

The total leakage of the local leak rate tests being performed on containment isolation components not previously tested, with three remaining to be tested, was in excess of 17.66 scmh (623.57 scfh). This value did not include any leakage from those components previously tested, nor did it reflect the actual leakage through penetration X-22, which was listed only as greater than 17 scmh (600 scfh). As noted above, Technical Specification 4.7.A.2.f.1 established the limit for local leak rates to be 5.37 scmh (189.60 scfh). This limit was established to ensure containment integrity.

The licensee had initiated a licensee event report on July 5, 1994, to address the identification of penetrations that had not been tested as required by 10 CFR Part 50, Appendix J. The licensee stated that the causes would be addressed in a supplement to the report.

On the basis of the test results for the newly tested components, the inspectors concluded that the licensee had exceeded the Technical Specification limit for leakage to ensure containment integrity for an extended period without taking the required corrective actions. As such, this is identified as Example 2 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

2.1.4 Redundant Containment Isolation Barriers

The licensee inspected approximately 300 penetrations during the performance of Special Procedure 94-202. A number of those penetrations were found to lack redundant containment barriers.

The licensee identified some penetrations with both isolation valves located outside the primary containment. However, between the containment wall and the first isolation valve outside containment, there existed a single vent, drain, or test connection valve. Examples of this type of single barrier were Penetrations X-21, X-22, and X-2[illegible].

Some penetrations were identified by the licensee with only a single isolation valve outside of containment. Penetrations X-29E, X-30E/F and X-33E/F were examples.

Penetrations X-21E and X-209A/B/C/D had thermocouple wires routed in piping through the penetrations. On the outside of containment was an open valve, incapable of closing, with an unidentified sealant that could not be determined to be qualified. These penetrations were determined to have an unqualified barrier.

Appendix B to 10 CFR Part 50, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications . . . . 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."

Additionally, in accordance with draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the USAR, "[a]ll lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space [were] provided with at least two isolation valves (or equivalent) in series."

The inspectors identified the failure to have redundant barriers as Example 2 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.5 Classification of Primary Containment Isolation Barriers

The licensee identified that approximately 300 examples of components associated with containment penetrations were not classified as essential. Draft General Design Criteria, Criterion 1, July 1967, as stated in Appendix F to the USAR, required ". . . those systems and components of the station which [had] a vital role in the prevention or mitigation of consequences of accidents affecting the public health and safety [were] designed and constructed to high quality standards . . ."

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."

In addition, 10 CFR Part 50, Appendix B, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications . . . 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."

The licensee concluded that those components not classified as essential were designed, fabricated, and erected to quality standards that did not reflect the importance of the safety function to be performed in accordance with 10 CFR Part 50, Appendix B, Criterion III; General Electric Design Specification No. 22A1153, Revision 1; or Appendix F to the USAR.

The failure to design, fabricate and erect the containment isolation barriers to quality standards that reflected the importance of the safety function was identified as Example 3 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.6 Containment Penetration Inspections

The inspectors reviewed a number of primary containment penetrations previously inspected by the licensee. For those penetrations, the inspectors concluded that the licensee's inspection had been accurate and the marked-up drawings reflected the actual condition in the plant.

2.2 Motor-Operated Valve Issues

On December 20, 1993, as documented in NRC Inspection Report 50-298/93-29, Valve HPCI-MOV-MO17 failed to stroke. The licensee formed a problem resolution team to investigate that failure. The team issued a report on January 7, 1994, that documented the team's findings. Those findings were that the failure was the result of fiberglass fragments between the limit switch contacts. The team presented this report as the response to Nonconformance Report 93-270 in order to recommend corrective actions.

On March 14, 1994, as documented in NRC Inspection Report 50-298/94-09, excessive leakage was noted during the venting of piping between Valves RHR-MOV-MO25A and -MO27A. In this instance, the licensee determined that the problem was related to foreign material deposited on the valve seat after maintenance that breached the residual heat removal system boundary.

On May 27, 1994, the licensee reported that Valve MOV-MO16 was found "partially deenergized" after attempting to close the valve. The licensee's investigation led to the identification of "particles" stuck between the contacts of the torque switch.

On June 20, 1994, the licensee reported that Valve RHR-MOV-MO27B was not capable of passing its local leak rate test. At the time of this inspection, the licensee had not determined a root cause for the failure. The licensee had decided to move the primary containment isolation function from Valves RHR-MOV-MO25A(B) and -MO27A(B) to Valves RHR-CV-26CV(27CV), RHR-MOV-MO274A(B), and -MO25A(B). The licensee performed this change by use of a safety evaluation that was performed in accordance with 10 CFR 50.59.

2.2.1 Safety Evaluation Review

The inspectors reviewed the safety evaluation and found that the evaluation was thorough and in accordance with the requirements of 10 CFR 50.59. The inspectors noted that, in order to accomplish this change, the licensee had to change operating procedures to prevent the opening of either RHR-MOV-MO274 valve and to ensure that the motor operator will remain deenergized when the reactor coolant temperature was above 100°C (212°F). Another change to the procedures was that shutdown cooling could only be initiated when the reactor pressure was less than 344.7 kPa (50 psig). The inspectors concluded that the licensee's change of primary containment isolation boundary was adequately justified, and appropriate procedural controls were identified.

2.2.2 Limit Switch Compartment Cleanliness

During review of the licensee's actions related to the lack of cleanliness inside the limit switch compartments, the inspectors found that the licensee had proposed a completion date of September 1994 for the corrective actions related to the failure of Valve HPCI-MOV-MO17. The licensee had not entered the corrective actions into its tracking system.

This was a concern to the inspectors for two reasons. The lengthy amount of time allowed to pass before the corrective actions were due increased the chances for similar events to occur. In this case, a similar event did occur when Valve MOV-MO16 failed to operate properly. The other concern was the failure to timely incorporate the corrective actions into the tracking system to assure that management is provided with an appropriate status of corrective actions. The licensee had indicated that the failure to track was a backlog problem because of an administrative overload. In each case, a condition report had been issued and initial corrective action initiated. The inspectors were concerned that the licensee would have failed to perform these corrective actions without the NRC inspection into the motor-operated valve issues.

The licensee did form a condition resolution team to review the failure of Valve MOV-MO16. This team had not issued its report, therefore, the inspectors did not review the licensee's actions for that failure. The review of the licensee's actions is considered to be an inspection followup item (298/9414-03).

2.2.3 Analysis of Other Valve Concerns

The failures of Valves RHR-MOV-MO27A(B) presented other concerns. One concern was related to the control of foreign materials when systems were breached. The inspectors noted that corrective actions had not been approved for the March event when weld slag was determined to be the cause of the problem. When questioned by the inspectors, the responsible engineer stated that this issue had been given lower priority and, in essence, that there was a lack of personnel to ensure the corrective action process was timely. Another concern was that the licensee had not considered any interim actions to prevent foreign material to get into systems other than a memorandum to maintenance personnel informing them of recent problems and instructing them to be careful.

The inspectors concluded that management attention was warranted in the areas of foreign material exclusion and the corrective action programs. The corrective action program was considered to warrant the attention because of the fact that it had been implemented only recently and the inspectors noted these concerns.

2.3 Switch Calibration

The licensee identified that several instrument pressure switches in Racks 25-5 and -6, subject to drywell pressure, were isolated during the performance of the containment integrated leak rate tests performed in accordance with the ASME Boiler and Pressure Vessel Code. The licensee stated that these instruments were isolated because the licensee's staff thought the test pressure (approximately 400 kPa (58 psi)) would damage the instruments. Local leak rate testing had not been performed in lieu of opening the valves to the racks during integrated leak rate testing. On July 8, 1994, Surveillance Procedure 6.3.1.1.2, Revision 0, "Primary Containment Instrument

Local Leak Rate Tests," was issued to initiate local leak rate testing for these pressure switches. The pressure switches on Racks 25-5 and -6 include: 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. The pressure switches perform scram, containment isolation, and emergency core cooling system initiation upon receiving a drywell pressure signal of 13.7 kPa (2 psig) or greater. The licensee contacted the instrument vendor and was notified that the instruments could withstand the test pressure, but should be calibrated after the test to ensure there was no shift in the operating characteristics of the instruments. The licensee stated that these instruments would be calibrated after being subjected to the pressure of the containment integrated leak rate test. The failure to perform local leak rate tests is identified as Example 3 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

2.4 <u>(Closed) Unresolved Item 50-298/9403-01: Use of Single Manual Valve for Containment Isolation</u>

NRC Inspection Report 50-298/94-03 summarized the inspection conducted during January 2 through February 12, 1994. The report discussed the use of a single manual valve for containment isolation, which was determined to be an Unresolved Item (298/9403-01) pending additional NRC review. The valves in question were all manual operated vents, drains, or test connections; a total of ten valves were affected.

During this inspection, the inspectors determined that the ten valves, identified in the earlier inspection, had been modified by means of a maintenance work request. The modification consisted of adding either a cap or plug, which acted as a second barrier for containment isolation. This design philosophy was consistent with draft General Design Criteria 53, as stated in Appendix F to the USAR. All material used in the maintenance work requests were classified essential, and their certification and traceability were available.

The licensee submitted its response to NRC Inspection Report 50-298/94-03 by letter dated May 31, 1994. The response stated that the licensee was reconstituting the design basis for the primary containment system and would evaluate the issue within that task. In addition, the licensee advised that it was pursuing efforts to resolve NRC concerns involving the identification and control of manual primary containment isolation valves, or more appropriately the administrative control of the valve and cap/plug combination. The licensee stated that it planned to complete this effort by August 1994.

In addition to the ten valves identified in Unresolved Item 298/9403-01, additional manual vents, drains and test valves were capped or plugged in accordance with Maintenance Work Request 94-2978 and its supplement 94-3116. This was discussed as a part of the design changes in Section 2.1.1 of this report.

In accordance with draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the USAR, "[a]ll lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space [were] provided with at least two isolation valves (or equivalent) in series."

The ten single isolation valves without a second barrier were identified as Example 2 of the apparent violation of 10 CFR Part 50, Appendix B, Criterion III, identified in Section 2.1.4 (298/9414-01).

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

*R. Gardner, Plant Manager
*R. Godley, Manager, Nuclear Licensing and Safety
*G. Horn, Vice President, Nuclear
*S. Jobe, Acting Senior Nuclear Division Manager, Safety Assessment
*J. Lynch, Manager, Engineering
*E. Mace, Senior Manager, Site Support
*J. Mueller, Site Manager
*J. Sayer, Technical Assistant to Plant Manager
*R. Wilbur, Division Manager
*V. Wolstenholm, Division Manager, Quality Assurance

1.2 Other Personnel

*H. Berchert, Director, Division of Radiological Health, State of Nebraska
*J. Parker, Midwest Power
*R. Stoddard, Lincoln Electric System
*W. Turnbull, Midwest Power

1.3 NRC Personnel

*A. Beach, Director, Division of Reactor Projects
*L. Callan, Regional Administrator, Region IV
*P. Goldberg, Reactor Inspector, Engineering Branch
*C. Hackney, State Liaison Officer
*P. Harrell, Chief, Reactor Projects Branch C
*R. Kopriva, Senior Resident Inspector
*W. Walker, Resident Inspector

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

* Denotes personnel that attended the exit meeting on August 12, 1994.

2 EXIT MEETING

An exit meeting was conducted on August 12, 1994. During this meeting, the scope and findings of the inspection were reviewed. The licensee acknowledged the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

14-4 (13)

June 23, 1993

Docket: 50-298
License: DPR-46

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

SUBJECT: INITIAL SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP) REPORT

This forwards the initial SALP report (50-298/93-99) for the Cooper Nuclear Station. The SALP Board met on May 20 and June 15, 1993, to evaluate the licensee's performance for the period January 19, 1992, through April 24, 1993. The performance analyses and resulting evaluations are documented in the enclosed initial SALP report.

In accordance with NRC policy, I have reviewed the SALP Board's assessment and concur with their ratings, as discussed below:

Overall, licensee performance declined in several functional areas from the previous SALP evaluation. A large number of equipment problems occurred during the latter part of this appraisal period that were caused, in part, by the failure of licensee employees to aggressively pursue the root cause of potentially significant equipment problems and to assume effective ownership of systems and components. The problems were also caused by the willingness of licensee personnel to live with problems rather than thoroughly evaluate degraded or potentially degraded equipment issues. The Cooper Nuclear Station staff appears to be satisfied with working around these problems and, as a result, the licensee's problem resolution process and corrective action systems have been weak. Many of these equipment problems were long-standing, and the failure to self-identify and correct the problems are viewed as demonstrated fundamental weaknesses in the oversight and self-assessment functions. These concerns were most evident in the areas of Maintenance/Surveillance and Safety Assessment/Quality Verification and, as a result, these areas were assigned a rating of Category 3.

In Engineering/Technical Support, significant weaknesses were observed in problem resolution by the site engineering group. The board was concerned with the examples of insufficient rigor applied to the evaluation and resolution of identified problems. The evaluations relied heavily on verbal information and there was lack of formality in the approach to the resolution of these problems which contributed to escalated enforcement actions. The board assigned a rating of Category 2 because of the performance of the corporate engineering group and the improvements in operations training.

N/1
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Performance in the functional area of Operations was mixed and assigned a rating of Category 2. Routine operations remained strong, but there was a lack of a questioning attitude on the part of the operating staff for some engineering operability determinations. This lack of a questioning attitude may have contributed to some of the plant problems identified during this period. The relationship between the operations and training staffs has improved but requires some additional attention.

In Radiological Controls, performance has improved. The radiological controls staff has made major strides in improving the overall program. The board was concerned, however, with the apparent lack of aggressiveness in identifying radiological performance weaknesses. Nevertheless, overall performance was assigned a rating of Category 2 and was assigned an improving trend.

Recurring problems in the areas of offsite notification, emergency assessment, and decisionmaking tended to offset the improvements noted in the area of Emergency Preparedness. The failures to follow up on previously identified findings and the additional violations indicated a need for increased management attention. This area was assigned a rating of Category 2 with a declining trend.

The area of Security continues to be a strength and was assigned a rating of Category 1.

On the basis of the SALP Board's assessment, the length of the SALP period will be approximately 15 months. Accordingly, the next SALP period will be from April 25, 1993, to July 30, 1994.

A management meeting has been scheduled with you and your staff to review the results of the initial SALP report. The meeting will be open to the public and held at the Cooper Nuclear Station security building auditorium on July 9, 1993, at 10 a.m. Within 20 days of this management meeting, you may provide comments on and amplification of, as appropriate, other aspects of the initial SALP report.

Your written comments, a summary of our meeting, and the results of my consideration of your comments will be issued as an appendix to the enclosed initial SALP report and will constitute the final SALP report.

Sincerely,

James L. Milhoan
Regional Administrator

Enclosure:
Initial SALP Report
50-298/93-99

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D:DRP

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EECollins:df	JEGagliardo	HRood	JRoe	LJCallan	SJCollins
6/ /93	6/ /93	6/ /93	6/ /93	6/ /93	6/ /93

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6/ /93	6/ /93	6/ /93

*Previously concurred

INITIAL SALP REPORT

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE

INSPECTION REPORT

50-298/93-99

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

January 19, 1992, through April 24, 1993

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I. INTRODUCTION

The Systematic Assessment of Licensee Performance (SALP) program is an integrated NRC staff effort to collect available observations and data on a periodic basis and to evaluate licensee performance based upon this information. The program is supplemental to normal regulatory processes used to ensure compliance with NRC rules and regulations. It is intended to be sufficiently diagnostic to provide a rational basis for allocating NRC resources and to provide meaningful feedback to licensee management regarding the NRC's assessment of their facility's performance in each functional area.

An NRC SALP Board, composed of the staff members listed below, met on May 20 and June 15, 1993, to review the observations and data on performance and to assess licensee performance in accordance with NRC Manual Chapter 0516, "Systematic Assessment of Licensee Performance."

This report is the NRC's assessment of the licensee's safety performance at Cooper Nuclear Station for the period January 19, 1992, through April 24, 1993.

The SALP Board for Cooper Nuclear Station was composed of:

Chairman

A. B. Beach, Director, Division of Reactor Projects (DRP), Region IV

Members

J. W. Roe, Director, Division of Reactor Projects III/IV/V, Office of Nuclear Reactor Regulation (NRR)
S. J. Collins, Director, Division of Reactor Safety (DRS), Region IV
L. J. Callan, Director, Division of Radiation Safety and Safeguards (DRSS), Region IV
J. E. Gagliardo, Chief, Project Section C, DRP, Region IV
H. Rood, Project Manager, Cooper Nuclear Station, NRR
R. A. Kopriva, Senior Resident Inspector, Cooper Nuclear Station, DRP, Region IV

The following personnel also participated in or observed the SALP Board meeting:

J. L. Pellet, Chief, Operations Section, DRS, Region IV
T. F. Westerman, Chief, Engineering Section, DRS, Region IV
P. H. Harrell, Chief, Technical Support Staff, DRP, Region IV
I. Barnes, Chief, Technical Assistant, DRS, Region IV
B. Murray, Chief, Facilities Inspection Programs Section, DRSS, Region IV
D. B. Spitzberg, Emergency Preparedness Analyst, DRSS, Region IV
C. J. Paulk, Reactor Inspector, DRS, Region IV
E. E. Collins, Project Engineer, Project Section C, DRP, Region IV
W. C. Walker, Resident Inspector, Cooper Nuclear Station, DRP, Region IV

II. SUMMARY OF RESULTS

Overview

Performance in the area of plant operations was mixed. The plant operations staff performed its duties in a conservative manner during routine operations. Command, control, and communications within operating crews and within the operations department has improved but remains inconsistent. Management attention and oversight of routine plant operations was evident. There has been a lack of a questioning attitude by the plant operations personnel of operability determinations. The relationship between operations and training improved; however, the operations department appeared to not totally support and reinforce the training department's formal training program. The emergency and abnormal operating procedures still exhibited some weaknesses.

In radiological controls, management provided strong support. External radiation exposure controls were implemented effectively. Excellent programs were maintained in the radiation protection area. One enforcement action involved numerous operators and an operations supervisor that showed a lack of respect for the special work permit process. The licensee effectively implemented planning and preparation for the 1993 refueling outage. Excellent coordination existed between the radiation protection department and other departments and a strong as-low-as-reasonably-achievable (ALARA) program was maintained. Management has not been aggressive in identifying radiological performance weaknesses.

In maintenance and surveillance the licensee's preplanning and work practices were coordinated and well controlled, and their work item tracking system was excellent. The performance of maintenance activities was mixed, although communications and supervisory oversight were good. Maintenance of motor-operated valves was generally good, but there were weaknesses noted with the installation of terminal lugs. Weaknesses were found in the licensee's maintenance of the reactor building and safety-related check valves. Several licensee event reports were submitted during the appraisal period because of improper maintenance. Program procedures for control and scheduling of surveillance activities were controlled and explicit. Weaknesses were found in the adequacy of technical justifications to verify the operability of equipment when Technical Specification testing acceptance criteria had not been met. Weaknesses were also seen in the licensee's testing of the pressure isolation valves, secondary containment isolation valves, and manual valves needed for safe shutdown of the plant.

In emergency preparedness, improvements were observed in certain important performance areas. Recurring problems were noted, however, in the areas of offsite notifications and emergency assessment and decision making. These problem areas, combined with certain failures to promptly followup on findings affecting emergency preparedness, and the violations which were identified, indicate a need for increased management attention in this program area.

Performance in the security area continues to be excellent. The program was effectively managed by personnel within the security department. Upper management provided strong support for the security program. Excellent programs were noted in the areas of testing, maintenance, staffing, audits, and the response to audit findings.

In engineering and technical support, performance was good. The interface between corporate engineering and site engineering was effective. The overall process to control projects and design modification activities appeared to be very effective. The temporary modification process was found to be well implemented. Configuration management was found to be effective. The licensee's plant procedures were generally well controlled and technically adequate to perform the desired actions. Improvements were seen in training; however, licensed operator training continued to need management attention and priority. Significant weaknesses were observed in problem resolution, and several examples of a lack of rigorous problem resolution were seen. Examples of over-reliance on verbal information and informality were seen which directly contributed to escalated enforcement actions.

In safety assessment and quality verification the licensee implemented an effective operability determination and evaluation process and deficiency report process. While some problems were effectively resolved, others were not, continuing to show significant weaknesses in the licensee's approach to the resolution of issues. The causes for ineffective problem resolution included informality, apparent unquestioning deferment of corrective actions for generic problems, the absence of corrective action for those instances where explicit regulatory requirements did not exist, and poor personnel performance in bringing deficiencies to management's attention. The licensee has planned or implemented extensive initiatives to improve performance in problem resolution, however, the effectiveness of the licensee's initiatives to address personnel performance and personnel attitudes remains to be seen. The licensee's oversight and self-assessment activities were not always acceptable and will require additional management attention to assure that these activities provide management with the critical insights into the performance of the plant and the operating staff.

Functional Area	Rating Last Period (07/16/90 to 01/18/92)	Rating This Period (01/19/92 to 04/24/93)
Plant Operations	2	2
Radiological Controls	2	2*
Maintenance/Surveillance	1	3
Emergency Preparedness	2	2**
Security	1	1

Engineering/Technical Support	2	2
Safety Assessment/ Quality Verification	2	3

*I Improving Trend - Licensee performance was determined to be improving during this assessment period. Continuation of the trend may result in a change in the performance rating.

**D Declining Trend - Licensee performance was determined to be declining during this assessment period and the licensee had not taken meaningful steps to address this pattern. Continuation of the trend may result in a change in the performance rating.

III. CRITERIA

The evaluation criteria, category definitions, and SALP process methodology that were used, as applicable, to assess each functional area are described in detail in NRC Manual Chapter 0516, dated September 28, 1990. This chapter is available in the Public Document Room files. Therefore, these criteria are not repeated here but will be presented in detail at the public meeting to be held with licensee management.

IV. PERFORMANCE ANALYSIS

A. Plant Operations

1. Analysis

This functional area consists primarily of the control and execution of activities directly related to operating the plant.

Evaluation of this functional area was based on routine inspections performed by the resident inspectors. The Region-based inspections included two operator examinations, two Emergency Plan inspections, one plant procedures inspection, and one unannounced followup inspection to observe licensed operators' conduct during in-house requalification examinations.

The previous SALP report (NRC Inspection Report 50-298/92-99) noted that management's attention and oversight was not always conservative; procedures were not always used properly; and that significant weaknesses were identified in the command, control, and communications activities when the operating staff was presented with simulated nonroutine emergency events.

Command, control, and communications within operating crews and within the operations department has improved but remains inconsistent. A training guide and an operations directive have been issued in this area. However, formal training to implement the guide and directive had not been provided and none of the on-shift supervisors questioned shortly after its issuance were aware of the operations directive. Management expectations and reinforcement of training in these areas is an ongoing challenge. For example, operations management was not expeditiously informed by a shift crew (by written or oral communications) that a problem with the control room annunciator computer resulted in 60 annunciators being in an alarmed condition. Control room logbook entries for the event were also unclear.

The last SALP report cited weaknesses in event diagnosis and implementing emergency and abnormal procedures effectively. During this SALP period, these problems appear to have been effectively addressed as indicated by improved diagnosis and procedure use during operator license and requalification examinations and emergency preparedness exercises and inspections. The last SALP also described concerns related to emergency and abnormal procedure validity. During this SALP period, the licensee was cited for the failure to incorporate changes reflecting plant modifications into the emergency support procedures in a timely fashion. This could have resulted in the procedures being unusable during certain accident sequences involving the release or potential release of radioactive material. This indicates that procedure implementation continues to be of concern, although for reasons different than described in the previous SALP.

The enforcement history in this functional area involved the failure to incorporate changes into the emergency support procedures and the failure to follow procedures, which resulted in a loss of shutdown cooling. The procedure violations were not repetitive of those addressed in the previous SALP report but are indicative of the fact that procedure implementation continues to be of concern.

While the licensee has implemented significant effort to formalize and document the evaluation of the immediate impact of deficiencies on the operability of systems, there has been a lack of a questioning attitude by plant operations of operability determinations prepared by engineering. Examples included the operability determinations that were prepared to address a temporary strainer in the suction of the reactor core isolation cooling system, leaking shutdown cooling suction valves pressurizing the low pressure residual heat removal system, and particulate contamination in emergency diesel generator fuel oil above the limits specified by the station procedures. In each case, the conclusion of operability was accepted without challenge. The operability determination for the temporary strainer contained assessments that the strainer could be back-flushed, but the physical configuration precluded back-flushing and no procedures existed telling operators how to perform the evolution. For the leaking valves, a vent path was established to bleed the pressure, but no limits were specified identifying how much leakage would be considered unacceptable, and no evaluation of the containment isolation function was made. For the high particulate, the condition was accepted without an evaluation of the impact of the deficiency on the fuel delivery system and the operability of the emergency diesel generator. The acceptance of these operability determinations with apparent weaknesses shows an absence of a questioning attitude and a lack of ownership by plant operations.

Management attention and oversight of routine plant operations was evident. Senior site management routinely toured the control room on a daily basis and, during major evolutions and/or plant changes, management personnel were present in the control room, providing on overview of the activities.

Management's actions in response to operational events were usually appropriate. On two occurrences the licensee elected to shut down the plant to implement corrective actions (replace batteries in April 1992 and repair the motive power to the low pressure coolant injection valves in September 1992). The licensee also made a decision to reduce reactor power after the design basis reconstitution group identified a problem with the control power for some emergency core cooling system valves.

The plant operations staff performed its duties in a conservative manner during daily, routine, steady-state power operations; reactor startups; and plant shutdowns. Few plant operational problems or perturbations were experienced during the reporting period, and the actions taken by the operators in response to a feedwater transient and reactor recirculation pump trip were accurate and timely. There were no automatic plant trips during this assessment period.

Observed communications between operating staff and other departments during the performance of maintenance and surveillance activities have improved from those observed in the previous SALP period. Managements' efforts had been successful in reducing the number of illuminated annunciators on the main control room boards during steady-state operations.

The relationship between operations and training improved. However, the operations department appeared to not totally support and reinforce the training department's formal program. Instances were noted where more emphasis was given to on-crew input into training content than to that prescribed by the formal training program. This may account for the differences identified in crew performance. Some cross-crew normalization progress has been made by rotating operators into the training department; however, the full benefit of the program has not been realized.

The licensee's operations staff was a very experienced and knowledgeable group of licensed senior reactor and reactor operators. During this assessment period, the licensed operator staffing remained adequate to maintain a six-shift rotation of operating crews.

Housekeeping in the plant was good. Most of the areas have been painted and have been provided adequate lighting. Labeling has been completed for most components throughout the plant and found to be of a quality to support component manipulations by plant personnel. There remain some less-trafficked areas in the plant, which are not up to the housekeeping equivalence exhibited by the majority of the plant areas.

In summary, overall performance in the area of plant operations was mixed. The plant operations staff performed its duties in a conservative manner during routine operations. Command, control, and communications within operating crews and within the operations department has improved but remains inconsistent. Management attention and oversight of routine plant operations was evident. Although different, the emergency and abnormal operating procedures still exhibited some concerns identified in the previous SALP report. There has been a lack of a questioning attitude by plant operations of operability determinations. The relationship between operations and training improved, however, the operations department appeared to not totally support and reinforce the training department's formal training program.

2. Performance Rating

The licensee is considered to be in Performance Category 2 in this functional area.

3. Recommendations

a. NRC Actions

Review the licensee's actions and training with respect to operator communications during nonroutine operating activities. Review the licensee's actions to enhance their operability determination process.

b. Licensee Actions

Licensee management needs to take appropriate measures to assure that the long- term issue of operator communications during nonroutine operating activities has been included in the training process for all operators. The licensee should implement an effective process for the evaluation of deficient conditions that impact the safe operation of the facility.

B. Radiological Controls

1. Analysis

This functional area consists primarily of activities related to radiation protection, radioactive waste management, radiological effluent control and monitoring, water chemistry controls, radiological environmental monitoring, and transportation of radioactive materials.

This area was inspected seven times by Region-based radiation specialist inspectors and on a continuing basis by the resident inspectors.

During the previous assessment period, concerns were identified involving the implementation of the radiological protection program during outages and routine, day-to-day activities. During this assessment period, the licensee improved implementation of the radiological protection program during routine, day-to-day activities, but still experienced some problems during outages when activity levels were high.

Enforcement was taken when several plant operators did not follow the requirements of a special work permit requirement. This example was of particular concern because numerous operators and an operations supervisor were involved. This event reflected a lack of respect for the special work permit process as an essential part of the radiation protection program.

Senior management's support for the radiation protection program, and the radiological protection management's oversight of day-to-day activities, was excellent. Strong programs had been developed and were maintained in the areas of control of radioactive materials and contamination, surveys, monitoring, and radiation instrument calibration.

Management has not been aggressive in identifying radiological performance weaknesses. During this assessment period, the licensee generated only five radiological safety incident reports. Given the number of plant areas that are contaminated and the magnitude of work performed, the absence of incident reports reflects a site attitude of not documenting, and consequently not aggressively pursuing, radiological problems.

Communications among the radiation protection department and other departments were instrumental in the progress made to reduce the number of contaminated areas within the radiological controlled area. The licensee planned to implement a program for controlling radiation exposures, which included a new radiological support system that used a state-of-the-art computer-based electronic dosimetry system and access control system.

The licensee effectively implemented planning and preparation for the 1993 refueling outage. The strengths of this program included an inventory of radiation protection supplies and equipment, coordination between the radiation protection department and other departments, and an appropriate number of contract radiation protection personnel to provide the required radiation protection coverage of outage activities. The contract technicians were brought on site several weeks prior to the outage to receive training.

External radiation exposure controls were implemented effectively by monitoring whole body exposures using thermoluminescent dosimeters, self-reading dosimeters, radiation surveys, radiation work permits, and administrative dose limits. Radiation areas and high radiation areas were properly posted and controlled. Special work permits were improved to provide enhanced guidance to workers and make them easier to understand. Isolated examples were noted of workers not following all of the instructions of special work permits. The licensee had implemented a good internal exposure control program.

The licensee had implemented an excellent ALARA program. The radiological protection department was proactive in the area of ALARA briefings, which were conducted prior to the performance of complex maintenance and operational activities and/or when the potential for high radiation exposure was present. The ALARA prejob briefings were thorough and well organized, addressed all important issues, and emphasized good radiological protection practices.

Prior to the 1993 refueling outage, the plant utilized a "soft" shutdown, which provided good control of crud bursts and improved reactor water cleanup, reducing external exposure. The ALARA suggestion program received an increase in ALARA suggestions and was given excellent support from management and workers from other departments. ALARA personnel performed daily reviews of the doses accrued by jobs during the 1993 refueling outage and made frequent tours of the drywell to observe work activities. Person-rem exposures and personnel contamination events were maintained below outage goals.

The licensee's liquid and gaseous radioactive waste effluent program, water chemistry and radiochemistry programs, and radiological environmental monitoring program were effective and well managed. The sampling results from all these programs compared well with NRC independent measurements.

The solid radwaste and radioactive materials transportation programs included excellent procedures for the preparation and shipment of radioactive waste and other radioactive materials. The licensee's performance of characterizing, classifying, and preparing radioactive waste for shipment and burial during this assessment period was excellent. Radioactive materials and waste shipments were made without incident or problems.

Staffing was maintained at appropriate levels in the radiological controls areas. The various departments in the radiological controls areas had experienced a very low turnover of technical personnel. The radiation protection staff was supplemented with contract radiation protection technicians during outages, but reliance was not placed on contractor personnel during normal operating periods.

Accredited training and qualification programs were established and being implemented for personnel in this functional area. The radiological controls area personnel were well trained and qualified. Training instructors were well qualified. Coordination existed between the training department and the various departments that received training in this functional area. The licensee's overall training efforts were excellent.

The quality assurance audits and surveillances performed in the radiological controls area identified pertinent findings, and the corrective actions for the findings were timely and comprehensive. The audit teams included qualified auditors and technical specialists who were knowledgeable of the applicable requirements to be reviewed in specific program areas. A self-assessment of the radiation protection program, including source term reduction, work control, communications, radiation protection during outages, ALARA, and training, was performed, and the assessment identified several recommendations for program improvement.

In summary, management provided strong support for the radiological controls area. External radiation exposure controls were implemented effectively. Excellent programs were maintained in the radiation protection area. One enforcement action involved several operators and an operations supervisor that showed a lack of respect for the special work permit process. The licensee effectively implemented planning and preparation for the 1993 refueling outage. Excellent coordination existed between the radiation protection department and other departments, and a strong ALARA program was maintained. Management has not been aggressive in identifying radiological performance weaknesses.

2. Performance Rating

The licensee is considered to be in Performance Category 2 in this functional area with an improving trend.

3. Recommendations

a. NRC Actions

None

b. Licensee Actions

The licensee needs to implement measures to assure that the facility staff is more aggressive in the pursuit of issues which are to be documented in the radiological safety incident report process established by site procedures.

C. Maintenance/Surveillance

1. Analysis

This functional area consists of activities associated with the predictive, preventive, and corrective maintenance of plant structures, systems, and components. This area also includes the conduct of surveillance testing, inservice testing, and inspection activities.

NRC inspection efforts consisted of routine inspections by the resident inspectors and five inspections performed by region-based inspectors. In the last SALP report, no recommendations were made for the overall program improvement.

During this assessment period, maintenance work practices were performed in a coordinated controlled manner. One exception to procedure compliance was observed during emergency diesel generator maintenance where workers did not obtain a system engineer inspection as required by the work package. The licensee continued to have an excellent work item tracking system, which is effective in assuring that work in progress is properly documented and work needing to be performed is prioritized appropriately.

The licensee's performance in implementation of maintenance activities was mixed. Preplanning of maintenance activities and attention to detail by maintenance personnel were good with good communication between maintenance personnel in the field and other organizations. Supervisory personnel presence was noted during complex activities and periodically during the performance of more routine efforts.

Maintenance of motor-operated valves was generally good. Some weaknesses were seen, however, in the maintenance of motor-operated valves. Discrepancies involving improper terminal lug installations and evidence of corrosion and dirt in the limit switch compartment for environmentally qualified motor-operated valves were not identified or corrected by maintenance personnel.

In mid-1992, the licensee initiated the development of a formal check valve program based on NRC and industry recommendations. A significant weakness existed, however, in the licensee's check valve maintenance and testing activities. While many check valves were tested in the inservice testing program and others were inspected by the preventive maintenance program, reactor coolant pressure isolation check valves were neither disassembled for inspection nor leak rate tested. The licensee's maintenance and testing activities did not ensure that these valves were capable of performing the safety-related pressure isolation function. At the end of the assessment period, the licensee was implementing plans to perform leak rate testing of these check valves.

During the refueling outage, testing of the secondary containment showed that the licensee had not effectively tested or maintained secondary containment. The secondary containment integrity test did not effectively address adjacent building status, and this masked identification of a significant deficiency. Also, features such as secondary containment isolation valve timing were not effectively tested. The licensee had not effectively maintained door seals, which were worn from use during the operating cycle, degrading the secondary containment. At the end of the assessment period the licensee was implementing corrective actions to address these deficiencies.

During this assessment period, safety-related systems were declared inoperable and licensee event reports were issued as a result of ineffective, or lack of, maintenance on plant equipment. The instances involved: (1) the clogging of a steam trap, due to a lack of preventive maintenance, that raised questions about the operability of the reactor core isolation cooling system, (2) inoperability of a damper in the control room heating and ventilation system because the linkage was not routinely lubricated, (3) failure of a motor-operated valve to operate due to a stripped stem nut on the valve which was not detected because of the lack of appropriate acceptance criteria in the maintenance work procedure, and (4) failure of a battery charger to operate properly due to a lack of preventive maintenance.

The systems engineering organization was involved in maintenance and surveillance activities. The oversight provided by the engineers helped to ensure that the maintenance and surveillance activities were acceptably implemented. However, the issues discussed in the four preceding paragraphs indicate shortcomings in program technical definition and technical resolution of identified problems.

Early in the assessment period, a significant weakness was found in the licensee's surveillance test program involving the station batteries. The program allowed that safety-related equipment could be considered operable without an adequate technical justification when Technical Specification test acceptance criteria were not met. Following identification of this issue, the licensee effectively implemented corrective actions to ensure that Technical Specification test acceptance criteria reflected actual operability criteria and that test discrepancies were formally evaluated and approved.

Program procedures for control and scheduling of surveillance activities were controlled and explicit. There were very few missed or overdue surveillance tests. The surveillance schedule consistently reflected planning and assigned priorities. Procedures for conducting surveillances were well written and easy to follow.

Personnel conducting surveillances were qualified. Senior technicians and senior operations personnel provided oversight and guidance to trainees while conducting on-the-job training. During surveillance performance, the licensee's staff continued to demonstrate good communication and coordination.

The performance of nondestructive examinations in the inservice inspection program was observed to be good. The nondestructive examinations were performed by contract personnel that were well qualified for the specific processes. The repair and replacement program was effectively implemented by well-documented work packages, and the performance of work activities was observed to be good.

The scope of the inservice inspection program did not include all safety-related heat removal systems, such as the service water and reactor equipment cooling system. These systems consequently have not received all the inspection activities specified by the Technical Specifications, including pressure testing. The licensee's third party review of the inservice inspection program did not identify these systems as needing to be included in the inservice inspection program.

The licensee's testing did not include periodic verification of many manual valves that were specified to be operated, using emergency operating procedures, or would need to be operated in other emergency conditions. One example was the emergency diesel generator fuel oil storage tank cross-connect valve.

A weakness was seen in the licensee's primary containment leak rate testing program. The licensee had tested 26 containment isolation valves with test pressure applied in a direction opposite to containment pressure without an adequate basis that the test results would be equivalent or conservative. Licensee testing with the test pressure applied in the direction of accident pressure demonstrated, for some valves, that the testing was nonconservative. At the end of the assessment period, the licensee was implementing corrective actions to either test the valves in the direction of accident pressure or provide an adequate justification that testing in the reverse direction was equivalent. The licensee also did not verify that instrumentation cabinets that would be exposed to primary containment pressure after the accident were tested. The hydrogen/oxygen analyzers were not tested at accident pressure.

In summary, the licensee's preplanning and work practices were coordinated and well controlled, and their work item tracking system was excellent. The performance of maintenance activities was mixed, although communications and supervisory oversight were good. Maintenance of motor-operated valves was generally good, but weaknesses were noted with the installation of terminal lugs. Weaknesses were found in the licensee's maintenance of the reactor building and safety-related check valves. Several licensee event reports were submitted during the appraisal period because of improper maintenance. Program procedures for control and scheduling of surveillance activities were controlled and explicit. Weaknesses were found in the adequacy of technical justifications to verify the operability of equipment when testing acceptance criteria had not been met. Weaknesses were also seen in the licensee's testing of the pressure isolation valves, secondary containment isolation valves, and manual valves needed for safe shutdown of the plant.

2. Performance Rating

The licensee is considered to be in Performance Category 3 in this functional area.

3. Recommendations

a. NRC Actions

The NRC should conduct inspection activities with the focus of assessing the technical adequacy of activities and the appropriate scope of activities and to review maintenance and surveillance program identification and resolution of conditions adverse to quality.

b. Licensee Actions

The licensee should review the scope and depth of maintenance/surveillance activities to make sure that the maintenance and surveillance programs for safety-related equipment are adequate to assure that the equipment can and will continue to perform its safety functions. The licensee should also increase the emphasis on oversight by plant management and systems engineering to provide an increased level of technical support to the maintenance and surveillance activities at the plant. Management should provide additional emphasis on generation of thorough and detailed maintenance and surveillance procedures, and on the need for maintenance/surveillance personnel to carefully follow the procedures.

D. Emergency Preparedness

1. Analysis

This functional area includes activities related to the establishment and implementation of the emergency plan and implementing procedures, onsite and offsite plan development and coordination, support and training of emergency response organizations, licensee performance during exercises and actual events that test the emergency plans, and interactions with onsite and offsite emergency response organizations during planned exercises and actual events.

The previous SALP report noted a Performance Category 2 in the emergency preparedness area. The report recommended licensee action to implement proactive corrective actions for identified weaknesses and to enhance its self-assessment capabilities.

Evaluation of this functional area was based on the results of two inspections conducted by the regional emergency preparedness analyst and observations by the resident inspectors. The two inspections included evaluation of the 1992 emergency exercise and an operational status inspection which included a regional inspection initiative to evaluate the knowledge and performance of duties of emergency response personnel.

During the assessment period, there were six emergency declarations associated with actual events, all at the Unusual Event classification level. Five of the declarations were made following initiation of a shutdown required by Technical Specifications. The sixth declaration was made following a minor earthquake detected onsite.

During two of these events, the licensee experienced some difficulties in implementing portions of the emergency plan and implementing procedures. Specifically, following one event there was a delay in event classification, which indicated a weakness in the decisionmaking process. In addition, a violation was cited for the licensee's failure to complete notifications to offsite authorities in a timely manner following the declaration of this event. Following a subsequent Unusual Event declaration, notification of one offsite organization was untimely. The licensee identified the problems noted above and initiated corrective action. In one instance, however, the licensee's process of investigating, formulating, and documenting the needed corrective action was slow.

The 1992 exercise resulted in five NRC identified weaknesses. The weaknesses involved: (1) weak analysis and technical assessment of plant conditions, (2) failure to take steps to ensure habitability of the Technical Support Center/Operational Support Center, (3) failure to detect and classify General Emergency conditions promptly, (4) failure to make the offsite notification of the General Emergency in a timely manner, and (5) use of multiple dose assessment programs for decisionmaking purposes without clear guidance on reconciling conflicting results. The weakness concerning analysis and technical assessment of plant conditions was found to be a repeat of a similar weakness identified during the previous exercise. During the exercise, the NRC noted licensee improvements in several areas from the performance in previous exercises. Most notable were improvements in the performance of control room operators, tracking of response teams, and the licensee's selfcritique process. The 1992 exercise was not evaluated by FEMA, however, the licensee demonstrated an excellent working relationship during the exercise with the state response organizations that participated.

As a result of the 1992 exercise weaknesses and the previously mentioned findings related to actual event declarations, a management meeting was held with the licensee to discuss NRC concerns in emergency preparedness.

The operational status inspection found that emergency response facilities had been well maintained. A good program of emergency response training had been administered and a good number of trained personnel had been assigned to the emergency response organization. Quality assurance audits of emergency preparedness were of good scope and depth. During emergency preparedness walkthroughs, operating crews performed well and demonstrated an improved knowledge and performance of duties in all areas found to be weak in recent inspections.

Two violations were identified during the operational status inspection. One violation was for failure to conduct required tests of the pagers used to notify members of the emergency response organization. The second violation was identified for failure to conduct a drill critique and for failure to follow up as required on drill weaknesses. A noncited violation was identified and corrected by the licensee for failure to submit to NRC one emergency plan implementing procedure revision within the required time frame.

In response to NRC recommendations from the previous SALP report, the licensee formed an emergency preparedness task force to review and recommend actions in areas such as emergency preparedness program effectiveness, the emergency plan, command and control of the emergency response organization, emergency preparedness training, exercises and drills, and other programmatic areas. The task force report was issued midway through the SALP period. Substantive recommendations and initiatives were made by the task force. Additional corrective actions and improvement initiatives were presented during the October 1992 emergency preparedness management meeting with the licensee. Many of the corrective actions and improvement initiatives arising from these efforts were scheduled for completion beyond this SALP period. Therefore, the overall effectiveness of these actions had not been evaluated by the NRC. Despite these self-assessments and licensee identified recommendations, however, the NRC continued to identify instances where the licensee was neither aggressive nor proactive in response to some emergency preparedness findings during the SALP period.

In summary, during the SALP period, improvements were observed in certain performance areas important to emergency preparedness. Recurring problems were noted, however, in the areas of offsite notifications and emergency assessment and decisionmaking. These problem areas, combined with certain failures to promptly follow up on findings affecting emergency preparedness, and the violations which were identified, indicate a need for increased management attention in this program area.

2. Performance Rating

The licensee is considered to be in Performance Category 2 in this area, with a declining trend.

3. Recommendations

a. NRC Actions

Conduct an assessment to verify that the recurring problems of offsite notifications, emergency assessments, and decisionmaking have been corrected.

b. Licensee Actions

Licensee needs to take actions to assure that the recurring issues in offsite notification, emergency assessments, and decisionmaking have been corrected.

E. Security

1. Analysis

This functional area consists of activities associated with the security of the plant, including all aspects of access control, security background checks, safeguards information protection, and fitness-for-duty activities and controls. Evaluation of this functional area was based on the results of two security inspections performed by regional inspectors and observations made by the resident inspectors.

The previous SALP report identified the security area as a Performance Category 1 and did not include any specific recommendations.

Two violations of program requirements were identified during the SALP period involving the failure to maintain control of a visitor and the failure to change locks after termination of security guards for cause. Licensee management took prompt and effective action to correct the violations, identify the root causes, and strengthen procedures to prevent recurrence.

The security program was effectively managed. Plant and corporate security management personnel maintained an excellent knowledge of current industry trends by being actively involved in industry groups. Security management and the staff were well trained and qualified security professionals with an excellent understanding of nuclear plant security objectives.

The security system received excellent maintenance support. Instrumentation and controls technicians were provided to promptly repair or replace any security equipment that required corrective maintenance. Repairs were normally completed in a timely manner which, in turn, reduced the time spent by security officers on compensatory posts. The support and cooperation among security, plant maintenance, and the instrumentation and controls group was excellent and there was strong evidence of management's commitment to maintain a high quality and effective security program.

An excellent security reporting program had been implemented. The security event reports and reporting procedures were well understood by security supervisors and consistent with NRC requirements. The security staff conducted excellent analyses of security events, identifying trends and developing sound resolutions to problems.

The security organization was staffed with an appropriate number of personnel to ensure that the security program was properly implemented.

The security training program was administered by a well qualified full-time staff. The program was consistent with the requirements of the NRC-approved Security Force Training and Qualification Plan. Personnel training records were current and well maintained. Personnel were knowledgeable of their responsibilities and performed their duties competently. However, the training section did not have any training aids available for hands-on type training in the early part of the SALP period. For example, there were no simulated weapons or explosive devices to use during training on x-ray equipment or during bomb search tactics. The video film library, at the time, was limited to three or four recently acquired films. The licensed developed some additional training aids toward the end of the SALP period. However, the lack of training aids detracted from an excellent training program.

The submitted revisions to the Security Plan, the Security Contingency Plan, and the Security Training and Qualification Plan under the provisions of 10 CFR Section 50.54(p) were technically sound and reflected well-developed policies and procedures. Security personnel involved in maintaining program plans current were knowledgeable of NRC requirements and objectives.

A comprehensive annual audit of the security program was conducted by the licensee's quality assurance group. The audit team included an auditor with nuclear security experience from another power reactor utility. The audit was performance-based and very well documented. The security department implemented prompt and effective actions in response to the audit findings.

In summary, the licensee continues to maintain an excellent security program. The program was effectively managed by personnel within the security department. Upper management provided strong support for the security program. Excellent programs were noted in the areas of testing, maintenance, staffing, audits, and the response to audit findings.

2. <u>Performance Rating</u>

The licensee is rated as Category 1 in this functional area.

3. <u>Recommendations</u>

None.

F. Engineering/Technical Support

1. Analysis

This functional area consists of technical and engineering support for all plant activities. It includes all licensee activities associated with the design of plant modifications; engineering and technical support for operations; outages, maintenance, testing, surveillance, and procurement activities; and training and configuration management.

NRC inspection efforts consisted of routine inspections by the resident inspectors, four region-based inspections, and one structural audit team inspection. The inspection effort included team inspections to assess the motor-operated valve Generic Letter 89-10 program and engineering and technical support functions. Additionally, two sets of licensed operator examinations were administered at Cooper Nuclear Station.

The previous SALP report recommended that licensee management should implement actions to correct the ongoing concerns identified with the licensed operator training program. During this assessment, improvements were seen in training; however, licensed operator training continued to need management attention and priority, as previously discussed in the Operations functional area.

During this assessment period, a review of design modification activities was performed. The overall process to control projects and design modification activities appeared to be very effective, with a small backlog of work. Procedures to control design changes and modifications were found to be comprehensive and well written as were the plant modification packages. A great deal of conservative engineering effort was usually incorporated into the modification process.

The temporary modification process was found to be well implemented, and temporary modifications were not left in place over six months. Particular strengths were noted in the weekly audit performed by senior licensed operators and the use and control of temporary modification tags.

The interface between corporate engineering and site engineering appeared effective. There was a very stable engineering staff with a low turnover rate. Good morale was observed, and staffing levels appeared consistent with the workload. Engineering personnel were qualified and trained and their responsibilities defined. Of particular note was the emphasis on certification of system engineers as shift technical advisors. Engineering appeared to have good credibility and working relationships within the licensee's organization.

Configuration management was found to be effective. Although the licensee's design basis reconstitution process was found to be somewhat delayed, issues have been identified by this program which were promptly addressed.

The scope of the licensee's program to test motor-operated valves was consistent with Generic Letter 89-10 and was managed by knowledgeable personnel. During NRC reviews, a number of weaknesses were identified including calculations, use of design basis parameters, and testing. Additionally, the licensee had addressed the recommendation of Generic Letter 89-10 to evaluate and trend motor-operated valve failures but had not yet implemented the procedures. Inspectors observed the conditions of the valves to be very good. Overall, the licensee's motor-operated valve testing was good.

In the area of engineering, the licensee's plant procedures were generally well controlled and technically adequate to perform the desired actions. Examples of weaknesses in procedure support were noted, including a lack of independent verification of a calculation, providing timely procedure change information to plant operators and a lack of information in relay maintenance procedures. In one case, support procedures were known to be in error and timely corrective action had not been performed to correct the errors.

The licensee's program for the training of candidates for an operating license was determined to be adequate. One weakness was observed in the origin of learning objectives.

Actions to strengthen this program continued with the reallocation of resources to training, but at a slow rate. Enlarging the training staff through direct hiring and implementation of the program to bring in licensed operators from the operations department had a positive affect on the operations department's acceptance to training. Some improvement was noted in the formal communication process between the operations and training department management staffs.

Significant weaknesses were observed in problem resolution. One cause for ineffective problem resolution was informality and this has manifested itself as a tendency to rely on verbal information over documentation or plant records. Plant engineers relied on verbal information from maintenance personnel, without verification, that no temporary strainers existed in the system, in deference to the information that was on approved drawings that showed that strainers were installed. This verbal information was found later to be in error. Plant engineers also relied on verbal information regarding the existence of documentation that temporary strainers had been removed during preoperational or startup testing, even though the documentation that the engineer reviewed indicated the exact opposite. This was presented to the NRC as justification that temporary strainers had been removed and was later found to be incorrect; temporary strainers were, in fact, in the system.

Informality was also seen in the licensee's resolution of a secondary containment integrity test failure as discussed in maintenance and surveillance. A lack of rigorous resolution of a high particulate concentrations in the diesel fuel oil and leaking shutdown cooling suction isolation valves was also seen. The secondary containment was declared operable without a good understanding of the causes for the test failure and without action to prevent recurrence. The licensee subsequently found that a loop seal was missing causing a 10-inch flow path between the reactor building and the radwaste building.

Overall, the performance in this functional area was mixed. The interface between corporate engineering and site engineering was effective. The overall process to control projects and design modification activities appeared to be very effective. The temporary modification process was found to be well implemented. Configuration management was found to be effective. The licensee's plant procedures were generally well controlled and technically adequate to perform the desired actions. Improvements were seen in training; however, licensed operator training continued to need management attention and priority. Significant weaknesses were observed in problem resolution and several examples of a lack of rigorous problem resolution were seen. Examples of over-reliance on verbal information and informality were seen which directly contributed to escalated enforcement actions.

2. Performance Rating

The licensee is considered to be in Performance Category 2 in this functional area.

3. Recommendations

a. NRC Actions

None.

b. Licensee Actions

The licensee needs to resolve plant problems by correcting the root cause, with the objective of closing the issue with finality, rather than by using a quick-fix approach to mitigate the immediate symptoms. The licensee should put more thoroughness, formality, and attention to careful documentation into the process. The licensee should also give management oversight and/or system engineering function more emphasis, with more responsibility and authority for reviewing all aspects of a problem.

G. Safety Assessment/Quality Verification

1. Analysis

This functional area includes all licensee review activities associated with the implementation of licensee safety policies, including licensee activities related to amendment, exemption, and relief requests and other regulatory initiatives. In addition, it includes licensee activities related to the resolution of safety issues, safety committees, self-assessment activities, and the effectiveness of the verification function in identifying and correcting substandard or anomalous performance, in identifying precursors of potential problems, and in monitoring the overall performance of the plant.

NRC inspection efforts in this area consisted of the core inspection program, regional initiative inspections, and NRR program reviews. The previous SALP report identified a high threshold for initiating nonconformance reports and that the licensee was not proactive in identifying potential safety issues in this area. During this assessment period, the licensee expanded the corrective action program to capture those deficient conditions that did not rise to the threshold of a nonconformance report. The programmatic features appeared to be an improvement in that additional items were captured for resolution that would not have been documented under the previous program.

Problem resolution, however, continued to show significant weaknesses. While some problems were effectively resolved from a safety perspective, others were not addressed or evaluated with sufficient rigor to assure that potential safety issues were clearly brought to management's attention and subjected to the comprehensive corrective action which would correct the root cause and prevent recurrence of the problem.

Examples of effective problem resolution were the items identified from the licensee's design basis reconstitution efforts, such as a single failure vulnerability in the emergency core cooling systems and the vulnerability of safety-related switchgear to missiles. In these examples, the licensee's understanding of the safety implications of the vulnerabilities was good, and the licensee implemented effective compensatory/corrective actions to resolve the problems.

Problems which were not adequately resolved included copper contamination in station batteries, temporary startup strainers in safety-related systems, repetitive feedwater check valve leak rate test failures, primary coolant system relief valve drift problems, informal documentation of deficiencies in emergency condensate storage tank inspections, emergency diesel fuel oil high particulate, leaking shutdown cooling suction valves, reactor building surveillance test failures, and, emergency operating support procedures with previously identified deficiencies that were not corrected.

The apparent causes for ineffective or protracted problem resolutions included: (1) apparently unquestioning deferment of corrective actions until the "generic" or "industry" problems have been solved; (2) reluctance to take corrective action in those cases where explicit regulatory requirements did not exist; and (3) reluctance by working-level personnel to bring problems to the attention of plant management.

The licensee's protracted resolution of feedwater check valves that failed local leak rate testing repetitively and the absence of action to prevent recurrence or to mitigate the primary coolant system relief valve setpoint drift are examples of a willingness to defer corrective action until generic issues are resolved. The licensee's operability conclusion for emergency diesel fuel oil high particulate and their ineffective initial corrective actions for leaking shutdown cooling suction isolation valves are examples of a reluctance to take corrective action without explicit regulatory requirements. The emergency condensate storage tank coating blistering which was found during an inspection, but not documented in the work package, was an example of the type of problem not brought to management's attention.

Plant management has shown the ability and desire to effectively resolve issues once they are made aware of the deficiencies. However, management continues to be, for the most part, reactive in identifying deficient conditions. Historically, the licensee had established a performance indicator which placed an uppor limit on the number of open corrective action documents. This was viewed as a reward for a low number of corrective action system documents and may have discouraged the documentation of deficient conditions. The initiation of nonconformance reports, historically, has been linked to reportability and/or operability. This fostered the practice of documenting only reportable conditions in the corrective action systems rather than documenting deficient conditions and then giving them the appropriate review for reportability. Deficiencies identified when equipment was not operable, or not required to be operable, were not likely to be captured by the licensee's corrective action systems. The licensee's initiatives in implementing a deficiency report process, while very positive, have not yet corrected the attitudes that remain from the historical approach to corrective action systems.

At the end of the assessment period, the licensee had taken corrective actions to improve performance in resolving problems, many of which had not yet been implemented. The licensee's programmatic initiatives appear sound; however, the effectiveness of the licensee's corrective actions to address personnel performance and personnel attitudes have not yet been evaluated.

Licensee efforts have also been expended to develop and implement formal operability determination and evaluation processes. These efforts were initiated in response to an operability determination which did not receive approval from the Station Operations Review Committee as required. The licensee had generally been effective in evaluating the immediate impact of deficient conditions on the operability of safety-related equipment, but the immediate conclusion of operability may have encouraged delay of prudent corrective actions in some cases. Also, some operability determinations contained weaknesses as discussed in plant operations.

The licensee's performance of oversight and critical self-assessment activities were marginally satisfactory. The Station Operations Review Committee and the Safety Review and Audit Board met frequently to evaluate emerging safety issues and to review other issues required by their charters and the Technical Specifications. The oversight activities of these committees had not been effective in identifying the numerous problems which were found by the NRC inspectors in the special strainer inspection and in the corrective action inspection.

Although the quality assurance department issued quarterly trend reports that contained a comprehensive compilation of activities, the reports did not highlight problems or provide any assessment or recommendations as a result of indicated trends. The audit and surveillance activities of the quality assurance department had not been effective in providing effective oversight of site activities to provide early identification of many of the issues that were identified in the special inspection on strainers and the corrective action inspection.

Station performance indicators had received limited distribution and did not contain an assessment of the indicators or draw conclusions that would have been of benefit to management in their oversight of site activities.

The licensee's system for identifying and evaluating internal and external operational experience and events had been effective as a management tool. The Document and Event Review Committee actions to identify training work requests for improving training effectiveness based on operational experiences was a strength.

During the assessment period, the NRR staff reviewed a large number of license amendment requests and the safety analyses performed by and for the licensee. Generally, the licensee's submittals were acceptable. The number of licensing actions and activities appears to be appropriate for a plant of Cooper Nuclear Station's vintage. Overall, the licensee's performance for this element of this functional area is average and could be improved by increased attention to timeliness, accuracy, and completeness. The licensee's performance has been good, however, when it focussed its resources on an issue. An example of this is the well-thought-out comments the licensee submitted regarding the staff's draft position on the generic dedication issues that resulted from the pilot inspections.

In summary, the facility has generally been operated in a safe manner. While some problems were effectively resolved, others were not, continuing to show significant weaknesses in the licensee's approach to the resolution of issues. The causes for ineffective problem resolution included informality, deferment of corrective actions for generic problems, the absence of corrective action for those instances where explicit regulatory requirements did not exist, and poor personnel performance in bringing deficiencies to management's attention. The licensee has planned or implemented extensive initiatives to improve performance in problem resolution, however, the effectiveness of the licensee's initiatives to address personnel performance and personnel attitudes remains to be seen. The licensee's oversight and self-assessment activities were not always acceptable and will require additional management attention to assure that these activities provide management with the critical insights into the performance of the plant and the operating staff.

2. Performance Rating

The licensee is considered to be in Performance Category 3 in this functional area.

3. Recommendations

a. NRC Actions

Review the licensee's actions to enhance their process for performing critical self-assessments of their performance and providing more depth to their corrective action processes.

b. Licensee Actions

Licensee management needs to perform a critical assessment of their corrective action processes in light of the problems identified by the NRC and correct the process to assure that the process is meeting licensee and NRC expectations.

V. SUPPORTING DATA AND SUMMARIES

A. Major Licensee Activities

1. Major Outages

On February 10, 1992, the plant was shut down to replace degraded 250-volt battery cells. The plant was returned to full power on February 15.

On April 19, 1992, the plant was shut down to replace additional cells in 250-volt batteries. The plant was returned to full power on April 27.

On July 30, 1992, the licensee imposed a restriction of 90 percent power to assure emergency core cooling capability because of a single failure vulnerability. On September 11, 1992, the plant was shut down to implement a modification to eliminate the single failure vulnerability. The plant was returned to full power on September 15.

On October 1, 1992, the licensee experienced a recirculation pump trip and operated in single loop at 50 percent power. The plant was returned to full power on October 5.

On January 24, 1993, the licensee reached the all-rods-out condition and began end-of-cycle coast down. On March 5, 1993, the plant was shut down from about 80 percent power to begin the refueling outage. At the end of the assessment period, the plant was in the refueling outage with the core off-loaded.

2. License Amendments

Eleven licensing amendments were issued during this assessment period.

3. Major Modifications

During the current refueling outage, the licensee planned to: (1) install a hardened wet-well vent at Cooper Nuclear Station in response to Generic

Letter 89-16. (2) remove the rod sequence control system from the plant. and (3) remove the main steam line radiation monitor scram and containment isolation function from the plant.

B. Direct Inspection and Review Activities

NRC inspection activity during the assessment period included 40 inspections. Approximately 5190 direct inspection hours were expended. which did not include operator licensing examinations or contractor hours.

14-4 (18)
Commitment to Appendix J ✓

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

NEMAHA COUNTY, NEBRASKA

DOCKET NO. 50-298

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ISSUED: FEBRUARY 14, 1973

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failure of the main steamlines outside the containment or the turbine-condenser and because of the conservative nature of the staff's analysis of the dose consequences. Following our review and approval of the improved MSLIV surveillance test program, the appropriate portions of that test program will be included in the Technical Specifications.

6.2.3 Leakage Testing Program

The primary containment and components which will be subjected to containment test conditions were designed so that periodic integrated leakage rate testing can be conducted at peak calculated accident pressure and reduced pressures. We have reviewed the proposed test procedures for determination of the primary containment overall leakage, as well as penetration and isolation valve leakage, for both preservice and inservice containment leakage tests.

Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, have generally been designed with the capability of being individually leak tested at peak calculated accident pressure. Large hatches have been strengthened structurally to sustain the pressures of individual leak tests. Systems designed prior to the implementation of Appendix J, such as the control rod drive penetrations and standby liquid control system, do not have design provisions for individual leak tests; however, the normal functional testing of these systems ensure their operability and thence the necessary containment integrity.