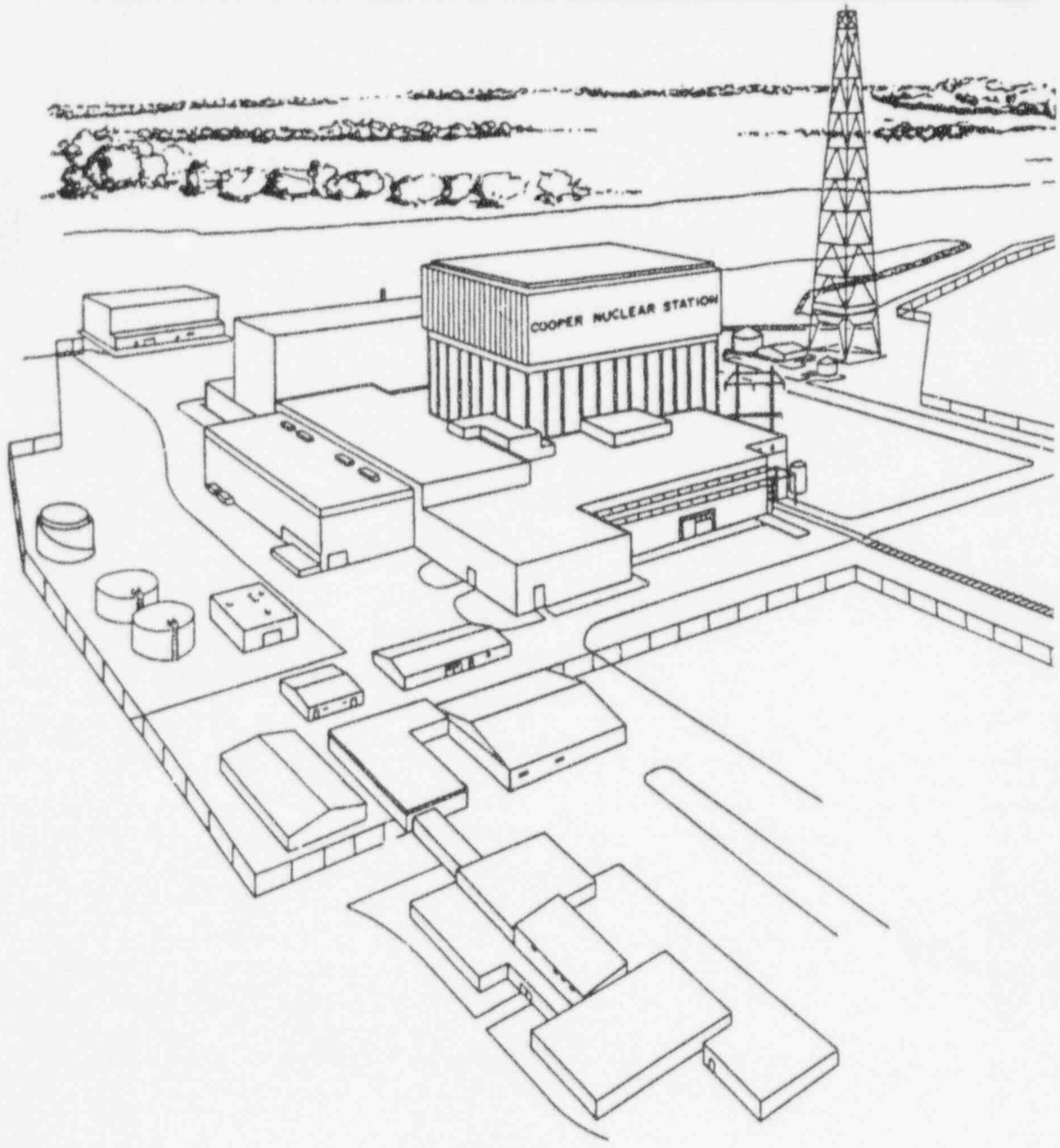
Cooper Nuclear Station

PERFORMANCE IMPROVEMENT PLANS
PHASE I: STARTUP PLANNING PROCESS



Nebraska Public Power District

August 25, 1994

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COOPER NUCLEAR STATION

PERFORMANCE IMPROVEMENT PLANS

PHASE I: STARTUP PLANNING PROCESS

Revision 0

APPROVED BY:

[signature] 8/25/94

Plant Manager Date

[signature] 8/25/94

Site Manager Date

COOPER NUCLEAR STATION

PERFORMANCE IMPROVEMENT PLANS

PHASE 1: STARTUP PLANNING PROCESS

Purpose

Cooper Nuclear Station is preparing a planning framework for lasting performance improvement at the station. The recent Diagnostic Self Assessment Team (DSAT) evaluation provided management with a set of performance issues to address that relate to both material condition items and program and process findings. There are a number of open issues including DSAT items, remaining Confirmatory Action Letter (CAL) items and inspection report open items, and CNS management-identified issues, including the Integrated Enhancement Plan (IEP). To assure that all issues are incorporated, NPPD management has determined that a more consistent and comprehensive planning process and framework needs to be established to guide the performance improvement efforts at CNS. This framework is in three phases:

1. Phase 1 - This phase is the startup phase and is the planning process that will address those significant issues identified in the DSAT, the remaining CAL and open inspection items, and CNS management-identified issues that must be resolved prior to plant startup.

2. Phase 2 - The short-term planning phase will be the first part of a process that is a building-block approach to performance improvement. This phase involves those essential management actions that will be accomplished in a short-term planning horizon (one to three months). Because this phase is of short duration, only a few of the highest priority issues will be addressed. Management believes that these issues are important to the station's near-term success and of such a nature as to lend themselves to rapid action.

3. Phase 3 - This is the long-term planning phase, and it will provide the framework for the management of the station's performance improvement actions essential to meet our long-term objectives for safety, production and economics. This phase is anticipated to constitute planning cycles from one to several years in duration.

This document is the Phase 1 Plan to provide the framework for near-term activities necessary fcr station restart. The plan discusses, where appropriate, interaction with the following two planning phases.

Phase 1 Objective:

Our objective is to identify the issues that must be resolved prior to the startup of the station to assure an error-free startup and a subsequent period of reliable operations.

Scope

The startup planning process will provide a comprehensive evaluation to assure that all significant issues for startup are resolved. These issues stem from three broad areas as follows:

1. Remaining CAL items and open inspection report items.

2. DSAT Open Items - Those issues, both hardware and program/process-related, that result from a thorough evaluation of the DSAT report.

3. CNS Management Issues - Hardware and other issues that CNS management is tracking for resolution prior to plant startup.

Methodology:

The process to identify and resolve the CNS startup issues will be conducted in three parts: (1) Issue Identification, (2) Issue Screening Evaluation, and (3) Issue Disposition. The potential effect of these issues on safe plant startup and continued operations will be evaluated to pre-established startup criteria. The issues will then be dispositioned for resolution prior to plant startup (Phase 1 of the performance improvement planning process) or the issue will be carried forward for future planning, resolution and closure (to phase 2, the short-term planning window or phase 3 in the long-term improvement plan for CNS). The characterization of each part is as follows:

Issue Identification

Issue Identification will involve identification of remaining CAL items and open inspection report items, DSAT open issues, and CNS management-identified issues. Identification of issues will be coordinated by a team of senior managers that will provide a high degree of assurance that all relevant issues are identified.

Once the complete set of issues is identified, the management team will bin them into either hardware issues or program and process issues. The program and process issues will be further characterized in specific areas to allow more effective issue evaluation and provide the ability to evaluate the full significance of the issues as they relate to the overall effectiveness of programs and processes at the station. The program and process areas are as follows:

1. Management - management development, leadership, and change management.
2. Independent Oversight and Self Assessment - roles and responsibility of SRAB, SORC, QA and QC, and organizational self assessment.
3. Corrective Action Program - problem identification, root-cause analysis, planning and issue resolution, performance monitoring and follow-up.
4. Operational Experience Review (OER).
5. Plant Testing - IST, surveillance, post-maintenance testing, preconditioning.
6. Work control - identification, tracking, planning and scheduling.
7. Configuration Management - clearance program, valve lineups, replacement parts, and drawing control.
8. Design Control - plant design change control.
9. Procedural Control - technical quality, procedure changes.
10. Engineering Support - roles, responsibilities, support to operations and maintenance.

Issue Screening Evaluation

Once the issues have been categorized, they will then be evaluated by CNS managers to determine if they should be resolved prior to station startup or carried forward for inclusion in the short-term or long-term planning phases. The issue screening evaluation process provides a formal method to assure each issue is addressed appropriately.

The Issue Screening Evaluation will be performed in two levels to pre-established criteria. This will allow station management to focus on those issues that are clearly important to plant restart, yet assure that all issues are captured for future resolution.

Level 1 Screening Evaluation - Issues are evaluated to identify potential safety issues or operability issues. These issues are automatically categorized as requiring resolution prior to plant startup.

Level 2 Screening Evaluation - The second level evaluation will characterize the remaining identified issues to determine if they should be resolved prior to startup. The following criteria will be used:

(1) An event, component failure, deficiency or condition that could result in operation in an LCO Action Statement.

(2) Failing to perform a required surveillance test or other license requirement or meet a commitment to an outside agency.

(3) Failure of power production equipment that could result in a plant transient, derate, or plant shutdown.

(4) Conditions that have resulted in repetitive safety system equipment failures.

(5) Potential licensing-basis deficiencies requiring maintenance to restore to conforming conditions, i.e., deficiencies in safety-related or other qualified equipment (e.g., EQ, Appendix R, or seismic).

(6) Potential design basis deficiencies, i.e., deficiencies in safety related equipment or other Technical Specification equipment not in conformance with the USAR.

(7) Deficiencies in configuration management programs, processes, engineering analysis codes, or documentation that have, or could have, a reasonable probability of affecting equipment operability.

(8) Conditions that may create an unacceptable potential for an unplanned radioactivity release to the environment or discharge of effluent in excess of limits.

Following the initial issue evaluation, the management team will have identified those individual issues that are required to be resolved prior to station restart. In addition, the management team will assess each of the program and process areas in an integrated manner, such that the effects of the deficiencies within each area are assessed in total.

Issue Disposition

Issue Disposition will take items that are identified as requiring resolution prior to plant startup and categorize them into three types: (1) Equipment/Hardware deficiencies, (2) Documentation (e.g., procedural, drawing, specification) deficiencies, and (3) Program and Process deficiencies. Each type of issue will be dispositioned as follows:

1. Equipment/Hardware - Resolved prior to startup by restoration to fully conforming status or, if acceptable, by approval of an engineering evaluation. In the latter case, full resolution after unit restart will be scheduled.

2. Documentation - The documentation will be revised and appropriate training conducted, or other compensatory measures will be taken if justified (e.g., issuing interim, controlled drawings or providing additional training). Any additional actions for full resolution will be scheduled after startup.

3. Program and Processes - Revise the program or process and conduct training, make interim changes to alleviate the deficiency, or take other compensatory measures (e.g., assign additional technical resources to work control to research work instructions). In the case of interim measures, a schedule for completion will be provided.

Review of Post-Startup Dispositioned Issues

The station management team will also review the remaining, non-startup items to confirm that the aggregate effect of the issues does not create the potential for problems at the station. This review will determine the cumulative effects of any work backlogs on station safety and reliability.

Independent Management Assessment

Following the completion of this plan to identify, screen and disposition required startup activities, the results will be independently reviewed by an appropriately established team to provide an assessment of the acceptability of the results to meet the objectives of this plan.

Draft NOV input
8/18/94

9414-01

10 CFR Part 50, Appendix B, Criterion III, states, in part, 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 Cooper Nuclear Station Updated Safety Analysis Report, Appendix F, "Conformance to AEC General Design Criteria," sates, in part, the ". . . the purpose of this appendix [is] to show that the design and construction of the Cooper Nuclear Station has been performed in accordance with these general design criteria."

Contrary to the above, Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," contained safety-related isolation valves but was not included on the safety-related drawing list as of July 1, 1994.

9414-02

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

Contrary to the above, as of May 14, 1994, the licensee failed to provide for Type C local leak rate testing of 68 components passing through 54 containment penetrations.

9414-03

10 CFR Part 50, Appendix B, Criterion III, states, in part, 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."

Draft General Design Criteria, Criterion 53, July 1967, in accordance with Appendix F to the USAR, states that "[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."

1. Contrary to the above, as of May 14, 1994, many penetrations were identified without redundant valving. These penetrations included, but were not limited to, penetrations X-21, X-22, X-25, X-29E, X-30E/F, X-33E/F, X-209A/B/C/D, and X-218.

2. Contrary to the above, as of February 22, 1994, ten manual operated vents, drains, or test connections had single manual valves for containment isolation.

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9414-04

10 CFR Part 50, Appendix B, Criterion III, states, in part, 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."

Draft General Design Criterion 1, in accordance with Appendix F to the Updated Safety Analysis Report, states 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 . . ."

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

Contrary to the above, the licensee failed to design, fabricate and erect approximately 300 containment penetrations to the standards specified in USAS B31.7-1969.

9114-05

Technical Specification 4.7.A.2.f.1 states, in part, that "local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves . . . The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La."

Contrary to the above, as of July 11, 1994, the total leakage for the valves and penetrations that had never been tested, with three tests remaining, exceeded the 0.60 La limit allowed by Technical Specifications. The 0.60 La limit was 5.37 scmh (189.60 scfh) and the leakage for the valves that had never been tested was in excess of 17.66 scmh (623.57 scfh).

Draft NOV input
8/31/94

9414-01

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

1. The Cooper Nuclear Station Updated Safety Analysis Report, Appendix F, "Conformance to AEC General Design Criteria," sates, in part, the ". . . the purpose of this appendix [is] to show that the design and construction of the Cooper Nuclear Station has been performed in accordance with these general design criteria."

   Contrary to the above, Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," contained safety-related isolation valves but was not included on the safety-related drawing list as of July 1, 1994, and some safety-related components were not included on the drawing.

2. Draft General Design Criteria, Criterion 53, July 1967, in accordance with Appendix F to the USAR, states that "[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."

   1. Contrary to the above, as of May 14, 1994, many penetrations were identified without redundant valving. These penetrations included, but were not limited to, penetrations X-21, X-22, X-25, X-29E, X-30E/F, X-33E/F, X-209A/B/C/D, and X-218.

   2. Contrary to the above, as of February 22, 1994, ten manual operated vents, drains, or test connections had single manual valves for containment isolation.

3. Draft General Design Criterion 1, in accordance with Appendix F to the Updated Safety Analysis Report, states 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 . . ."

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   Contrary to the above, the licensee failed to design, fabricate and erect approximately 300 containment penetrations to the standards specified in USAS B31.7-1969.

A/28

9414-02

Technical Specification 4.7.A.2.f.1 states, in part, that "local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves . . . The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La."

1. Contrary to the above, as of May 14, 1994, the licensee failed to provide for Type C local leak rate testing of 68 components passing through 54 containment penetrations.

2. Contrary to the above, as of July 11, 1994, the total leakage for the valves and penetrations that had never been tested, with three tests remaining, exceeded the 0.60 La limit allowed by Technical Specifications. The 0.60 La limit was 5.37 scmh (189.60 scfh) and the leakage for the valves that had never been tested was in excess of 17.66 scmh (623.57 scfh).

3. Contrary to the above, several instrument pressure switches had not had local leak rate testing performed after being isolated from the containment integrated leak rate test.