Omaha Public Power District
444 South 16th Street Mall
Omaha NE 68102-2247

February 7, 1997
LIC-97-011

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555-0001

References: 1. Docket No. 50-285
2. Letter from NRC (J. M. Taylor) to OPPD (F. M. Petersen), "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," dated October 9, 1996

SUBJECT: Response to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information

On October 11, 1996, Omaha Public Power District (OPPD) received the Reference 2 letter, which required submittal of a response under oath or affirmation within 120 days of receipt. Accordingly, this letter with its attachments constitutes the required response from OPPD; this information is intended to provide the U. S. Nuclear Regulatory Commission (NRC) added confidence and assurance that Fort Calhoun Station is operated and maintained within the plant design bases, with any deviations identified and reconciled in a timely manner.

The following information is provided to facilitate review of this submittal. The response format consists of this transmittal letter with 3 attachments. Attachment 1 is an affidavit; Attachment 2 includes the responses to the requested information items (A) through (E), including descriptions of design review and reconstitution efforts. Attachment 3 is a listing of Acronym Definitions and Referenced Procedures.

The responses also contain references to important milestones in the history of Fort Calhoun Station (FCS). The Safety System Outage Modification Inspections (SSOMI) were conducted by the NRC in late 1985, as documented by Inspection Reports 85-22 dated January 25, 1986 and 85-29 dated March 19, 1986. The subsequent SSOMI reinspection was documented by Inspection Report 88-200 dated September 16, 1988. These inspection reports identified several violations and other deficiencies associated with maintenance of the FCS design bases and processes used to design, install, and test modifications. These and other problems led to FCS being placed on the list of plants requiring additional NRC attention. OPPD commissioned an Independent Nuclear Assessment (INA) by an engineering consultant to identify areas needing improvements.

In order to best administer implementation of the numerous corrective actions resulting from the SSOMI and the INA, OPPD created the Safety Enhancement Program

(SEP) in 1988. In the development of the SEP, detailed analyses of problems in approximately 19 areas were performed. This work included the assessment of root causes, evaluation of actions required to resolve major concerns, and the determination of action priorities based upon their contribution to the safety of FCS. The SEP included 72 groups of actions, including major items such as reconstitution of the FCS design bases, procedure upgrades, and improvement of configuration control processes. Both internal audits and NRC inspections verified the completion and effectiveness of the corrective actions. In 1989, FCS was removed from the list of plants requiring additional NRC attention.

Various administrative controls are discussed in the attached responses. Quality Procedures (QPs) for the Nuclear Operations and Production Engineering Divisions provide overall administrative guidance for major programs and processes. QPs assist in ensuring that the identified activities are conducted in accordance with OPPD policies, practices, and regulatory requirements, including the FCS Quality Assurance Plan. QPs generally define "who, what, and when." Production Engineering Instructions, including General Engineering Instructions, implement QPs by generally providing guidance on "how," i.e, approved standards or methods for performing engineering activities. The FCS Operating Manual contains Standing Orders, the Radiological Emergency Response Plan, and various types of procedures and instructions, arranged and numbered in functional areas such as Operations, Radiation Protection, Chemistry, and Maintenance.

The attached responses contain discussions of various enhancements and continuing corrective actions. None of these discussions are considered new or different commitments to the NRC, with the following exception:

> In order to provide additional assurance of conformance with the design bases, OPPD is currently completing assessments of selected FCS systems using the Nuclear Energy Institute (NEI) *Guidelines for Assessing Programs for Maintaining the Licensing Basis* (NEI 96-05). Deficiencies found are entered into the Condition Report system for resolution. The assessment of the Chemical and Volume Control System is complete, and assessments of the Safety Injection and Instrument Air systems are ongoing. OPPD will complete assessments of all remaining FCS safety related and safety significant systems no later than February 1, 1999.

A significant example of a previously committed ongoing action is the biennial performance of internal Safety System Functional Inspections (SSFIs) on selected systems. The intent has been to validate the effectiveness of the Design Basis Reconstitution Program by assessing the ability of the systems to meet their design bases. Accordingly, OPPD has completed 9 SSFIs (including 1 re-inspection) on 8 different systems. These SSFIs have confirmed the operational readiness of the systems and the effectiveness of corrective actions implemented as part of the SEP.

The processes, procedures, and programs described in Attachment 2 provide reasonable assurance that the overall physical configuration of FCS is consistent

with its design bases, and that any deviations discovered are resolved in a timely manner using the corrective action program.

Please contact me if you have any questions.

Sincerely,

W. G. Gates

W. G. Gates
Vice President

Attachments

TCM/tcm

c: Winston & Strawn
F. J. Miraglia, Jr., Acting Director, Office of Nuclear Reactor Regulation
L. J. Callan, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector

LIC-97-011
Attachment 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Omaha Public Power District	)	Docket No. 50-285
(Fort Calhoun Station	)	
Unit No. 1)	)	

AFFIDAVIT

W. G. Gates, being duly sworn, hereby deposes and says that he is the Vice President responsible for nuclear activities at the Omaha Public Power District; that as such he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached Response to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information; that he is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge, information and belief.

W. G. Gates

W. G. Gates
Vice President

STATE OF NEBRASKA)
) ss
COUNTY OF DOUGLAS)

Subscribed and sworn to before me, a Notary Public in and for the State of Nebraska on this 7 day of February, 1997.

Carol J. Kelley

Notary Public

GENERAL NOTARY-State of Nebraska
CAROL J. KELLEY
My Comm. Exp. May 3, 1998

LIC-97-011
Attachment 2

Omaha Public Power District Responses to
NRC Request for Information Pursuant to 10 CFR 50.54(f)
Regarding Adequacy and Availability of Design Bases Information

Omaha Public Power District Response to NRC Request A

(A) Description of engineering design and configuration control processes, including those that implement 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR Part 50

Executive Summary

This response provides a description of the Omaha Public Power District (OPPD) engineering design and configuration control processes, including those that implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR 50. These processes are designed to provide reasonable assurance that Fort Calhoun Station (FCS) is operated and maintained in accordance with the design bases. The current programs and procedures were developed following the NRC Safety System Outage Modification Inspection (SSOMI) in 1985. They implemented improved engineering design and configuration control processes as part of the Safety Enhancement Program (SEP), in parallel with the Design Basis Reconstitution (DBR) Project. To ensure that activities such as modifications, procedure changes and safety evaluations could be conducted while the DBR Project was in progress, interim controls were established to ensure that design basis margins were not abrogated. These controls are discussed in the response to Request C.

Significant elements of the OPPD engineering design and configuration control processes are listed below.

1. Design work is reviewed for conformance with the design bases; if required, changes are made to the design bases. The Design Change Packages and supporting material (calculations, analyses, specifications, drawings, etc.) are developed using approved procedures and are maintained as controlled documents. (Section A.1)

2. System interaction analysis is used to assess the effects of a design change. The impact of a design change on plant procedures and training is assessed. Design verification is conducted in accordance with established procedures. Plant and Engineering management approval processes are defined. (Section A.1)

3. 10 CFR 50.59 requirements are implemented. Proposed changes to procedures and the plant design, including the USAR, are evaluated for 10 CFR 50.59 applicability. If applicable, Unreviewed Safety Question determinations are performed. The review and approval process for 10 CFR 50.59 evaluations is defined. (Section A.4)

4. 10 CFR 50.71 requirements are implemented. Changes conducted in accordance with 10 CFR 50.59 are reviewed to determine if USAR changes are required. NRC Safety Evaluation Reports are reviewed and appropriately included in the USAR. (Section A.5)

5. FCS personnel are trained and qualified to perform the processes and procedures for engineering design and configuration control (Section A.6).

6. To ensure conformance with the design bases for modifications designed and installed from initial commercial operation up to the Design Basis Reconstitution, OPPD conducted a safety evaluation check of Design Change packages (for safety related and nonsafety related modifications that had potential to impact safety systems). Concerns identified through this review were dispositioned using the DBD open item process. (Section C.2)

7. Assessments conducted since the SSOMI provide assurance that effective processes and procedures have been adopted. These audits, surveillances and inspections also confirmed that deficiencies are promptly corrected. The audits of current programs and ongoing trending of configuration control discrepancies confirm that the current processes are effectively implemented and are producing quality products. These trends also confirm the adequacy of the Safety Evaluation checks performed for modifications designed and installed before the Design Basis Reconstitution.

A.1. Engineering Design & Configuration Control Processes

The OPPD engineering design and configuration control processes for FCS are designed to ensure configuration changes and document changes are conducted in accordance with applicable rules and regulations including the requirements of 10 CFR 50 Appendix B Criterion III. These processes are depicted on Figure 1.

The following processes are used to make configuration changes to FCS plant systems, structures and components:

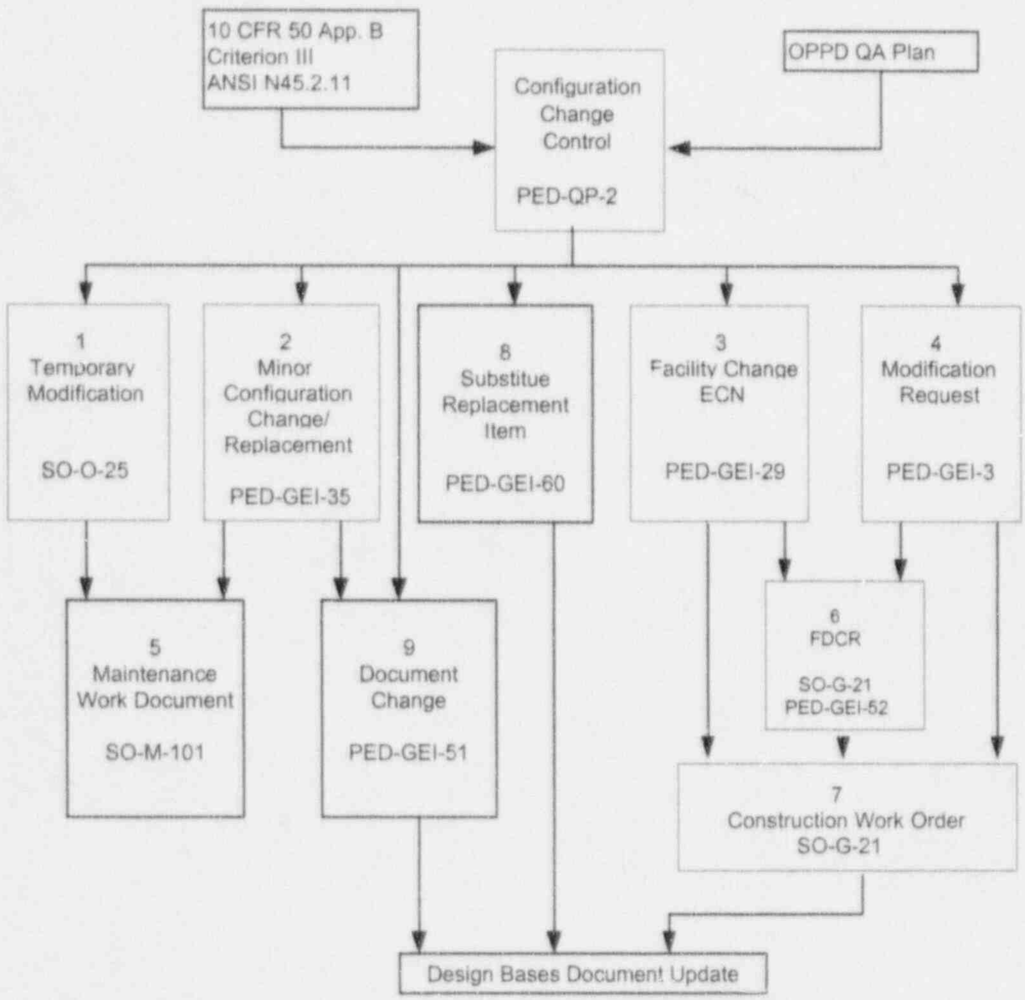
1. Modification Request (MR),
2. Facility Change Engineering Change Notices (FC-ECN),
3. Minor Configuration Change or Replacement (GEI-35), and
4. Temporary Modification (TM)

The Substitute Replacement Item Engineering Change Notice (SRI-ECN) and the Document Change Engineering Change Notice (DC-ECN) are processes used to revise design bases documents without changing the form, fit or function of the FCS systems, structures, and components. The procedures listed below and other documents provide detailed instructions for engineering design and configuration control at FCS.

1. Production Engineering Division Quality Procedure PED-QP-2, *Configuration Change Control*
2. Standing Order SO-G-21, *Modification Control*
3. Production Engineering Division General Engineering Instruction, PED-GEI-3, *Preparation of Design Change Packages*

4. Production Engineering Division General Engineering Instruction, PED-GEI-29, *Facility Change Evaluation*
5. Production Engineering Division General Engineering Instruction, PED-GEI-35, *Preparation of EARs for Minor Configuration Changes and Replacements*
6. Standing Order SO-O-25, *Temporary Modification Control*
7. Nuclear Operation Division Quality Procedure NOD-QP-3, *10 CFR 50.59 Safety Evaluations*
8. Nuclear Operation Division Quality Procedure, NOD-QP-16, *Updated Safety Analysis Report (USAR)*
9. Production Engineering Division Quality Procedure PED-QP-3, *Calculation Preparation, Review, and Approval*
10. Production Engineering Division Quality Procedure PED-QP-5, *Engineering Analysis Preparation, Review, and Approval*

Figure 1
Engineering Design and Configuration Control Processes



1. Temporary Modification: Short term alteration to the Plant that involve changes to the design bases.

2. Minor Configuration Change/Replacement
Simple changes that do not affect design parameters, operating conditions or functions

3. Facility Change ECN: Changes to non-CQE systems, structures or components that do not affect safety related design parameters, operating conditions, or functions

4. Modification Request: A change to the form, fit, or function of a CQE or Limited CQE system, struct... or component

5. Maintenance Work Document: Authorize, document and control maintenance work activities

6. Field Design Change Request: Field change to an approved design change package or ECN

7. Construction Work Order: Authorize, document and control construction work activities.

8. Substitute Replacement Item: Authorize and document the equivalency for the replacement of a part or component that does not have the same make, model and/or part number as the original.

9. Document Change: update of existing documents to reflect as-built conditions.

A.2. Configuration Changes

The configuration change processes used at FCS (Modification Request (MR), Facility Change Engineering Change Notices (FC-ECN), Minor Configuration Change or Replacement (GEI-35), and Temporary Modification (TM)) are described in the following sections.

Configuration Changes --Modification Requests

Standing Order SO-G-21, *Modification Control*, establishes procedural guidance for initiating, reviewing, approving and canceling Modification Requests. Modification Requests generally involve changes to the station that are complex from a system interaction perspective and affect system design parameters as well as operating conditions. The Modification Request process must be used for any configuration change that involves:

1. a change, directly or indirectly, to the fit, form or function of a CQE or a Limited CQE system, structure, or component,
   OR
2. a change to a Non-CQE system, structure, or component that affects a CQE or a Limited CQE system, structure, or component.

CQE designation corresponds to the more common industry designation "safety related." Limited CQE designation corresponds to the more common industry designation "Category II." The definitions of CQE and Limited CQE are:

> Critical Quality Element (CQE): Those structures, systems, components or items whose satisfactory performance is required to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.
>
> Limited Critical Quality Element (Limited CQE): Those structures, systems, components or items whose satisfactory performance is required to prevent or mitigate the failure of those structures, systems, components or items identified as Critical Quality Elements.

Modification Requests are documented in a Design Change Package (DCP) in accordance with PED-GEI-3. A DCP typically includes the following:

1. A description and evaluation of the problem addressed by the modification and the selected solution.
2. Any regulatory requirements associated with the modification. This includes the impact of the modification on the USAR, Technical Specification and licensing commitments.
3. Design Inputs

4. Design Analysis
5. Systems Interaction Analysis
6. 10 CFR 50.59 Safety Evaluation
7. Operating Impact
8. Installation and Testing Requirements
9. Drawing and Document Revisions

DCP Development -- Conformance with Design Basis

The information necessary to assess impact of a modification on the Design Bases is obtained from controlled documents including the DBDs, the USAR, drawings, specifications and other controlled design documents. This information is assembled in the Design Inputs section of the DCP. Using the Design Inputs and the guidance provided in PED-GEI-3, the conformance to the design bases of a modification is confirmed.

DCP Development -- Documentation of Supporting Analyses

Analyses and calculations to support a modification are included or referenced in the DCP. Cumulative changes are confirmed to be consistent with the design bases. Calculations are detailed technical packages that include the equations, references, data inputs, outputs, assumptions, and conclusions needed to substantiate engineering decisions. Calculations are prepared reviewed and approved using PED-QP-3. An Engineering Analysis (PED-QP-5) may also be used to prepare supporting analyses.

Engineering Analyses and calculations are reviewed and independently reviewed for analyses designated as CQE, Limited CQE and Fire Protection to provide design verification.

The review is performed by an individual technically qualified in the specific discipline or field of analysis. The analysis is reviewed for technical accuracy, completeness, and compliance with licensing commitments and procedure requirements. Independent reviews may be performed by one of three methods.

1. Analysis review
2. Alternate calculation
3. Qualification testing

DCP Development -- Systems Interaction Analyses

System Interaction Analyses are used to determine the impact of a modification on plant level design bases. (Plant level design bases documents are discussed in the response to Request C.) Checklists are used to screen a modification for the following potential system and plant interactions.

1. Fire Protection
2. Special Service Engineering Programs
3. Motor Operated Valve Program
4. Electrical Equipment Qualification
5. High Energy Line Break
6. Seismic Interaction, Qualification, and Effect on the USI A-46 Program
7. Electrical System Analysis
8. Human Factors Review
9. Security System
10. Environmental and Radiological Release
11. Materials Compatibility
12. Containment Integrity
13. Control Room Habitability
14. Missile Protection
15. Structural Impact
16. Independence Criteria
17. Single Failure Criteria
18. Possibility of Operator Error
19. Heavy Loads
20. Impact on HVAC
21. Use of Vendor Procedures
22. Station Blackout

This analysis confirms that the modification conforms to the design bases.

DCP Development -- Safety Evaluation (10 CFR 50.59)

The modification DCP contains the 10 CFR 50.59 Safety Evaluation. The Safety Evaluation addresses the three stages of the modification; Design, Installation and Testing. The safety evaluations are prepared and reviewed by engineers who are trained and qualified.

The Design Safety Evaluation ensures that the modification, as installed, will not introduce an unreviewed safety question. The safety evaluation also identifies the USAR changes that are needed as a result of the modification. (The USAR is later updated in accordance with procedure NOD-QP-16.)

The Installation Safety Evaluation addresses the specific configuration requirements, restrictions on interfacing equipment, systems and structures and the impact on the safety of the unit during construction.

The Testing Safety Evaluation assesses the modification testing requirements, including operational, functional and special tests, and associated test acceptance criteria. Details of the 10 CFR 50.59 safety analysis process are provided in Section A.4.

DCP Development -- Operating Impact

The impact of a modification on As Low As Reasonably Achievable (ALARA) exposure goals, Training, Testing, Operations and Maintenance is evaluated. The ALARA evaluation describes the impact of the modification on overall worker exposure during construction, and subsequent operation and maintenance of the components. The Training evaluation documents training program changes required due to design considerations or equipment selection. Impact on the Training Simulator is also addressed. The Testing evaluation describes any new or revised periodic testing requirements (e.g., changes to Inservice Inspection programs, etc.) for the modification. The Operation evaluation identifies changes to the operation of equipment including revision of operating procedures. The Maintenance evaluation describes any new or revised periodic maintenance requirements due to installation of the modification.

DCP Development -- Revisions to Procedures and Design Basis Documents

The DCP contains draft additions or revisions to the FCS Operating Manual procedures and documents. Drafts of the additions or revisions to the USAR, DBDs, and other design bases documentation are also provided in the DCP.

DCP Development -- Installation and Testing

The DCP provides a summary of the construction specifications that are required to install and test the modification. The package addresses the major construction aspects of the work and integration of plant outage and operational requirements for construction. Testing requirements are identified, evaluated and summarized. The test specifications include testing the component(s) and system to ensure functionality in actual operation. This may include any unique or special criteria for interruption and continuation of a test, requests for documenting, and/or review of unusual conditions.

The Construction Work Order (CWO) is used to authorize, document and describe construction work activities, including required QC inspections, for the installation of modifications. The CWO work instructions are developed from the installation and technical description sections of the DCP.

DCP Development -- Review, Approval and Acceptance

The DCP is reviewed to ensure technical adequacy of the modification. An Independent Review is performed for DCPs affecting CQE, Limited CQE, Fire Protection or Security systems or components. The Independent Review is performed by someone other than the preparer who is technically competent in the specific discipline or field of analysis.

The DCP is reviewed and approved by the Station Modification Acceptance and Review Team (SMART), a multi discipline team composed of personnel from Design and System Engineering, Operations and appropriate support groups.

Following SMART approval of the DCP, the DCP is presented to the PRC for review and recommendation of approval, in accordance with Technical Specification 5.5.1.6. In parallel, the Nuclear Safety Review Group (NSRG), on behalf of the Safety Audit and Review Committee, reviews the DCP in accordance with Technical Specification 5.5.2.7 to verify that the modification does not constitute an unreviewed safety question. Final approval of the DCP is by the Manager - FCS.

Following installation of the DCP, the following steps are completed to return the new or modified equipment to service:

1. Completion of installation and testing of a modification is verified by Engineering and Quality Control personnel.
2. Engineering personnel prepare the as-built drawings, panel schedules and electrical load distribution lists.
3. The appropriate System Engineer and Operations personnel update the Operating Manual procedures necessary for operation of the modified system.

Approval for return to service (Accepted for Operability) is authorized by the Manager-Operations, Manager-Maintenance, Manager-System Engineering, and the Shift Supervisor.

DCP Development - Field Changes

A Field Design Change Request (FDCR)is used to change a DCP after the DCP has been approved for construction. The FDCR documents the reason for the change and the documents affected. The affected documents are revised and attached to the FDCR. Engineering reviews the requested FDCR for conformance with scope, intent and design basis of the DCP. This includes a review of the 10 CFR 50.59 Safety Evaluation included in the DCP. The FDCR is screened to determine if Plant Review Committee (PRC) review is required. The FDCR requires PRC review and recommendation of approval to the Manager-FCS if the change reduces the ability of a system, component or structure to perform its design function, affects the conclusion of the 10 CFR 50.59 Safety Evaluation or significantly changes the modification scope. An FDCR not requiring PRC review is approved by the appropriate Engineering department head. After approval of the FDCR, the revised documents are incorporated into the original DCP. The Construction Work Orders are then revised to incorporate the FDCR.

DCP Development - Modification Close-out

Upon verification that all portions of a modification are completed, the Modification Completion Report is prepared to initiate the modification close-out process. This process includes the following steps listed.

1. Quality Control verifies that all construction and testing have been completed. Items verified are as follows:
   a. All CWOs have been completed and approved.
   b. All pre-operational testing has been completed and approved.
   c. All FDCRs have been reviewed, approved, and implemented.
   d. Independent Design Verification or Independent Review is complete.
   e. Pre-operational Tests have been completed.
   f. All changes to the construction drawings and drawing list have been documented by FDCRs.
   g. Construction and testing documentation is complete.
   h. All Condition Reports which could affect operability of the modification have been resolved.

2. The appropriate System Engineer verifies that the modificat: : has been installed, tested, and Accepted for Operability in accordance with the approved DCP and any FDCRs.

3. The construction drawings and the plant drawings are verified to ensure they reflect the as-built condition.

4. Labels are installed for all equipment affected by the modification.

5. Any additional revisions to the Operating Manual are completed. The documentation revised includes applicable:

   Abnormal Operating Procedures (AOP)
   Emergency Operating Procedures (EOP)
   Annunciator Response Procedures (ARP)
   Operating Procedures (OP)
   Operating Instructions (OI)
   Calibration Procedures (CP)
   Maintenance Procedures (MP)
   Preventive Maintenance Procedures (PM)
   Emergency Plan Implementing Procedures (EPIP)
   Radiological Emergency Response Procedures (RERP)
   Security Procedures
   Surveillance Test Procedures (ST)
   Standing Orders (SO)

6. The Vendor Manuals are updated and changes to Preventive Maintenance practices are made.

7. The affected System Training Manuals are revised.

8. The EEQ Program is updated.

9. The recommended spare parts are identified.

10. The ISI Program Plan is updated to reflect changes made by the modification.

11. Training materials are updated and training is conducted or scheduled.

12. The Maintenance Rule scoping documents are updated.

13. A PRC subcommittee reviews the DCP and close-out documentation to verify that the modification was correctly installed and tested, and that appropriate documentation is revised. Following completion of its review the PRC subcommittee recommends acceptance of the modification to the PRC.

14. The PRC accepts the modification.

15. Following acceptance, the PRC notifies the NSRG of their acceptance of the modification. The NSRG conducts a review of the Safety Evaluations.

16. The "as-built" DCP information is incorporated into the DBDs and other design documentation.

17. The DCP is processed for permanent retention.

Facility Change-ECN

The Facility Change ECN allows simple configuration changes to FCS equipment and supporting facilities. These changes:

1. Do not directly or indirectly affect CQE or Limited CQE systems, structures or components;
2. Do not affect the Technical Specifications;
3. Do not result in an Unreviewed Safety Question; and
4. Do not directly or indirectly affect Nuclear Safety.

Facility Change ECNs are documented in an ECN in accordance with PED-GEI-29. The FC-ECN provides the information necessary to construct and test the modification.

FC-ECN Development

The processes used to evaluate the conformance with design bases, conduct the systems interaction analysis, determine the operating impact, and determine installation and testing requirements for a FC-ECN are similar to those required to develop a MR. The processes are reduced in scope compared to the MR because a FC-ECN may not directly or indirectly affect nuclear safety.

FC-ECN Development -- Safety Evaluation (10 CFR 50.59)

Facility Change ECNs require a documented safety evaluation to ensure the configuration change does not result in an unreviewed safety question or require a Technical Specification change. This evaluation is performed in accordance with NOD-QP-3 and attached to the ECN. The safety evaluation considers all aspects of the Facility Change including design, installation and testing phases.

FC-ECN Development -- Installation and Testing

FC-ECNs are installed and tested via the CWO process in a manner similar to that used for Modifications.

FC-ECN Development -- Review, Approval and Acceptance

FC-ECNs are reviewed for technical adequacy by someone other than the Preparer competent in the specific discipline, or field of analysis. An Independent Review is performed for all FC-ECNs affecting Fire Protection, Radioactive Waste Disposal or Security. The Independent Review is performed by someone other than the Preparer who is technically competent in the specific discipline or field of analysis. An Engineering Department Head ECN review is also conducted as an overview to ensure commitments are met and the package is complete.

FC-ECNs are reviewed and approved by the Manager-FCS or Assistant Plant Manager-FCS prior to installation.

Completion of installation and testing of a FC-ECN is verified by Engineering. The close-out process for FC-ECNs is the same as that used for MRs.

Minor Configuration Change or Replacement

A Minor Configuration Change or Replacement is a change to the facility which does not alter the design, function, or method of performing the function of a component, system, or structure described in the Safety Analysis Report. A Minor Configuration Change or Replacement does not require detailed engineering and is simplistic in character.

To demonstrate a minor configuration change or replacement does not alter the design, function, or method of performing the function of a component, system, or structure described in the Safety Analysis Report, a 10 CFR 50.59 screening evaluation per NOD-QP-3 is required.

A Minor Configuration Change or Replacement is installed and tested using a Maintenance Work Document (MWD). Post Maintenance Testing requirements are specified within the MWD. The appropriate System Engineer reviews the MWD for adequate Post Maintenance Testing requirements. Documents that require updating as a result of the change (e.g., drawings, procedures, manuals, data base, etc.) are updated using a Document Change Engineering Change Notice.

Configuration Control During Maintenance Activities

Standing Order SO-M-101, *Maintenance Work Control*, identifies the methods for control of maintenance activities and defines the different methods of configuration control for maintenance work.

1. MWDs are verified to ensure that no configuration changes are proposed unless properly authorized.

2. Craft Personnel are responsible for verifying those components being replaced do not cause an unauthorized configuration change. Prior to beginning work on the maintenance work package, the craft personnel are responsible for comparing the parts and materials against approved drawings and/or design change documents to ensure that configuration control will be maintained.

3. The appropriate System Engineer is responsible for performing the technical review of maintenance work documents and ensuring that no unauthorized configuration changes are performed. The System Engineer also ensures adequate post maintenance testing is identified.

4. The Shift Supervisor is responsible for verifying the tag out boundaries such that the plant is operated in accordance with the Technical Specifications.

Temporary Modifications

Temporary Modifications (TM) are temporary minor alterations made to plant equipment that do not conform with original drawings or other design documents. These alterations are temporary in that they are expected to be installed for a short duration.

Temporary Modifications receive a screening for applicability of 10 CFR 50.59 and an unreviewed safety question determination if 10 CFR 50.59 is applicable. During this process, the design, installation and testing phases of each temporary modification are

evaluated. The technical evaluation of a temporary modification's impact is assessed with regard to items such as fire hazards, electrical loading, environmental qualifications, flooding, system interaction, and associated procedures.

Temporary Modifications are required to be reviewed and approved by the PRC prior to their installation. The Shift Supervisor may authorize installation prior to PRC approval in case of an emergency. Emergencies include situations when it is necessary to increase the margin of plant safety, prevent injury to personnel or to prevent damage to equipment. The Temporary Modification must be reviewed and approved by the PRC within two working days after installation. Temporary Modifications are installed using the MWO process. Temporary Modifications are identified on controlled drawings in both the Control Room and the Operations Control Center.

The use of a Temporary Modification for an extended period of time is minimized. Installed temporary modifications are reviewed by the PRC on a six-month frequency. Temporary Modifications that require an outage for removal are expected to be installed for only one operating cycle. Those that can be removed during plant operation are expected to be removed within six months of installation. Progress on removal of Temporary Modifications is tracked on a weekly basis through a report to management and monthly basis by a performance indicator. Operator awareness of installed temporary modifications is maintained by requiring the shift supervisor to review the TM log during shift turnover.

A.3 Design Document Changes Involving No Changes to Design Bases

A DC-ECN is used to authorize the evaluation and update of existing documents to reflect as-built conditions. An SRI-ECN is used to ensure equivalency of the replacement for a part or component that does not have the same make, model and/or part number as the original.

Document Change Engineering Change Notice -- DC-ECN

Document Change ECNs (DC-ECNs) are used when discrepancies are found between design and station documents or when field conditions do not match the design or station documents. DC-ECNs may only be used when the proposed change is substantiated by existing design basis documents and does not change the current station configuration. When document discrepancies are identified, they are evaluated by Engineering to ensure that the existing design basis supports the As-Built condition. If, based on that evaluation, the field condition is not substantiated by the existing design bases, the DC-ECN is canceled and the condition is documented and resolved via a Condition Report in accordance with Standing Order R-2.

DC-ECNs do not require a safety evaluation because there are no proposed changes or tests to the facility that could impact the Technical Specifications, USAR, or basis thereof, and/or result in an unreviewed safety question.

A review of a DC-ECN is conducted by an engineer other than the preparer. The appropriate Engineering department head approves a DC-ECN. Upon approval the appropriate changes are made to the documentation.

Substitute Replacement Item Engineering Change Notice -- SRI-ECN

An SRI-ECN is used to conduct and document an engineering equivalency evaluation of replacement parts or components that do not have the same make, model and/or part number as the original. This evaluation ensures that interface, interchangeability, safety, fit and function requirements are maintained in accordance with applicable regulatory or code requirements. The SRI-ECN Equivalency Evaluation performed per PED-GEI-60 demonstrates that the Substitute Replacement Item is being procured to an equivalent specification. If Equivalency cannot be demonstrated, then an SRI-ECN is not applicable.

Provided that equivalency of the SRI has been demonstrated, the component or part replacement will not result in a change to the facility as described in the USAR. Therefore, a 10 CFR 50.59 safety evaluation is not required.

The SRI-ECN documents that the Substitute Replacement Item is equivalent to the original item for all critical design characteristics. The SRI-ECN also documents that the design basis of the plant, system or component is not altered by the Substitute Replacement Item.

Each SRI-ECN is reviewed for technical adequacy by someone competent in the specific discipline or field of analysis and by someone other than the preparer. An Independent Review is performed for all SRI ECNs affecting CQE, LCQE, Fire Protection, Radioactive Waste Disposal or Security. The appropriate System Engineer reviews the SRI-ECN and performs an overall check to ensure the SRI will accomplish its objective. The Engineering department head reviews and approves the SRI-ECN.

The part or component approved by the SRI-ECN is installed using the MWD process discussed in Section A.2.

A.4 Safety Evaluations Implementing 10 CFR 50.59

10 CFR 50.59, Paragraph (a), Item 1, permits changes to FCS and its operation as described in the USAR without prior NRC approval, provided a change in the Technical Specifications is not involved or an Unreviewed Safety Question is not created. Proposed changes that are determined to involve a change in the Technical Specifications

or an Unreviewed Safety Question must be submitted to the NRC in accordance with 10 CFR 50.90 and 50.92 and 10 CFR 51.21.

Information from the NRC Inspection Manual, Part 9900, April 4, 1996; NRC I&E Manual, Part 9800, January 1, 1984; NRC Inspection Manual, Inspection Procedure 37001; NSAC-125; and NSAC-105 was used to develop the 10 CFR 50.59 Safety Evaluation program at FCS. The program is defined by Nuclear Operation Division Quality Procedure NOD-QP-3, *10 CFR 50.59 Safety Evaluations*.

NOD-QP-3 is the only procedure used at FCS for the preparation of 10 CFR 50.59 safety evaluations. The activities which must be evaluated for potential unreviewed safety questions in accordance with this procedure are:

1. All proposed procedures and proposed changes to procedures governing activities at FCS that could affect nuclear safety.

2. All proposed tests or experiments that could affect nuclear safety.

3. All proposed changes or modifications to plant systems or equipment that could affect nuclear safety.

NOD-QP-3 is comprehensive in its guidance:

1. The procedure presents a series of questions for review, followed by a detailed discussion of areas to be reviewed in responding to the question. The guidance provided in this document is structured to provide assurance that relevant aspects of the activity are considered by the preparer and identified for evaluation by the reviewers.

2. The procedure specifies the qualification and training requirements which must be satisfied by personnel preparing and reviewing nuclear safety evaluations.

3. The procedure provides the necessary direction for the accumulation of the information needed for a report to the NRC describing those activities performed under the provisions of 10 CFR 50.59, Paragraph (a), Item 1.

4. The procedure contains definitions of key words and phrases pertaining to the safety evaluation process to provide those individuals who are assigned preparation and review responsibilities with a consistent interpretation of their meaning.

The safety evaluation is a two step process. The first step is the 10 CFR 50.59 Applicability Screening; the second is the Unreviewed Safety Question Determination.

Screening to Determine Compliance with USAR and Technical Specifications

A guide is provided for documenting the 10 CFR 50.59 applicability screening process. This guide requires answers to the following series of questions.

1. What activity is being evaluated?
2. What is being done?
3. Why is this being done?
4. Does the activity involve a change to the Technical Specifications?
5. Does the activity being evaluated involve changes in the facility as described in the USAR? (NOD-QP-3 includes guidance on assessing interim changes not included in the USAR.)
6. Does the activity involve proposed tests or experiments not described in the USAR?
7. Could the activity adversely affect nuclear safety?

The Applicability Screening is used to determine if the activity affects the USAR, Technical Specifications, or License Conditions for FCS. When the responses on an Applicability Screening indicate that the change affects the USAR, an unreviewed safety question determination is initiated. If, however, the activity involves a change to the License Conditions or Technical Specifications, other than the basis section, the preparer is directed to process the change in accordance with Nuclear Operation Division Quality Procedure NOD-QP-7, *Facility License Changes (FLC)*. This screening ensures compliance with the requirements of both 10 CFR 50.59 and 10 CFR 50.92.

Unreviewed Safety Question Determination

The applicability screening determines which activities and modifications require evaluation of the unreviewed safety question criteria contained in 10 CFR 50.59, Paragraph (a), Item 2. The three criteria as stated in 10 CFR 50.59 have been expanded into several questions for clarity in responding to the requirement. The three criteria center on four main areas of evaluation:

1. Probability of an accident or equipment malfunction
2. Possibility of a previously unevaluated accident or equipment malfunction
3. Consequences of an accident or equipment malfunction
4. Reduction of the margin of safety

A guideline is provided for documenting the development of background information and the unreviewed safety question determination for the proposed activity. Completion of this documentation results in a positive or negative determination. Any unreviewed safety question identified must be submitted to the NRC for approval as part of a license amendment application.

Review and Approvals

Unreviewed safety question determinations are presented to and reviewed by the Plant Review Committee (PRC) and approved by the Manager-FCS. PRC makes a recommendation for approval or disapproval to the Manager-FCS.

All approved Safety Evaluations are reviewed by the Safety Audit Review Committee (SARC) through the Nuclear Safety Review Group (NSRG), acting on behalf of the SARC. The NSRG provides a bimonthly status report of this independent review and evaluation to the Safety Audit Review Committee (SARC).

A.5 10 CFR 50.71(e) USAR Updating Process

Procedure NOD-QP-16 provides assurance of compliance to 10 CFR 50.71(e). Revisions to the USAR are identified and documented during the 10 CFR 50.59 safety evaluation process and reviews of NRC Safety Evaluation Reports.

Identifying USAR Revisions for License Amendments and New Safety Issues

The USAR update procedure, in conjunction with the procedure for license amendments, requires that a review of NRC Safety Evaluation Reports be conducted to determine any changes to the USAR and Design Basis Documents. This ensures that "safety evaluations performed by the licensee in support of requested license amendments" are all reviewed and appropriately included in the USAR as required by 10 CFR 50.71(e).

Other SERs are reviewed for potential revisions to the USAR. This ensures that "all analyses of new safety issues performed by or on behalf of the licensee at Commission request," are appropriately reviewed for inclusion into the USAR.

A.6 Training

The Engineering Support Personnel (ESP) Training Program provides the training for those individuals who are engaged in engineering support functions at FCS.

The design and development of the instructional material used in the Engineering Support Training Program are based on National Academy for Nuclear Training document ACAD 91-017, *Guidelines for Training and Qualification of Engineering Support Personnel*. The Instructional Systems Design (ISD) process for performance-based training was utilized to implement the ACAD guidelines. The training program consists of two components: initial training, which includes orientation and position-specific training, and continuing training.

The objective of the Engineering Support Training Program is to provide engineers with job-related skills and knowledge to supplement their formal education. Initial training

provides technical knowledge of station operation related to the individual's job responsibilities. It also provides participants with the skills to perform assigned duties independently in a manner that promotes safe and reliable plant operations. Continuing training helps participants maintain and enhance their job performance and develop a broader scope and depth of knowledge and skills.

Maintenance personnel receive training on the requirements for configuration control as part of the training conducted on SO-M-101.

A.7 Audits and Inspections

Internal Quality Assurance (QA) audits have evaluated the engineering design and configuration control processes both during normal (SSFI type Audit) operation and during refueling (SSOMI type Audit) operations. Additionally, Quality Assurance personnel perform additional audits of both the Engineering Configuration Management program and the Station Engineering program. Several NRC inspections have also assessed the FCS engineering design and configuration control processes. Discussions of these audits and inspections is included in the responses to Requests C and E.

Omaha Public Power District Response to NRC Request B

(B) Rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures

Executive Summary

Omaha Public Power District (OPPD) processes, procedures, and programs provide reasonable assurance that Fort Calhoun Station (FCS) design bases requirements have been properly translated into operating, maintenance and testing procedures. This assurance is based on:

1. Completion in 1991 of an upgrade of safety related operating, maintenance and testing procedures to industry standards (Section B.1)

2. Development and use of a procedure writer's guide with requirements for incorporation of design bases information into operating, maintenance and testing procedures (Section B.1 and B.2)

3. Requirements for review of new procedures and procedure changes for conformance with the design bases (Section B.2)

4. Requirements for verification and validation of new procedures, along with selected verification and validation of procedure changes dependent on the complexity of the changes (Section B.2)

5. Access to and familiarity with the design bases by procedure writers and reviewers (Section B.2)

6. Overall positive experience with upgraded FCS procedures (Section B.3)

7. Lack of programmatic deficiencies in the procedures, their implementation or results, as identified by self assessment (Section B.3)

8. Results of inspections by the NRC of procedures, their implementation and results (Section B.3)

9. Additional verification of the procedures for conformance with the USAR and design bases currently being conducted by OPPD. (Section B.3)

B.1 Procedures Improvements

Procedures Upgrade Project

The primary objective of the Procedures Upgrade Project (PUP) was to upgrade operating, maintenance and testing procedures at Fort Calhoun Station (FCS) to current industry standards. Those standards included the following:

1. ANSI N18.7, *American National Standard Administrative Controls for Nuclear Power Plants*

2. Regulatory Guide 1.33, *Quality Assurance Program Requirements (Operation)*

3. NUREG/CR-1369, *Procedure Evaluation Checklist for Maintenance, Test and Calibration Procedures Used in Nuclear Power Plants*

4. INPO 85-026, *Writing Guideline for Maintenance, Test, and Calibration Procedures*

5. INPO 85-038, *Guidelines for the Conduct of Maintenance at Nuclear Power Station*

The PUP satisfied commitments associated with the FCS Safety Enhancement Program (SEP) Item 48. The scope of the PUP was to completely rewrite/upgrade the operating procedures, maintenance procedures and surveillance tests at Fort Calhoun Station. The term "upgrade," as used for the PUP, means to correct technical content deficiencies and human performance deficiencies, such that verbatim compliance with the procedure can be achieved by the least experienced but qualified individual. Approximately 3000 safety related and non-safety related procedures were upgraded by the PUP. The following procedures were specifically included:

1. Operating Instructions
2. Operating Procedures
3. Annunciator Response Procedures
4. Chemistry Procedures
5. Mechanical Maintenance and Pressure Equipment Procedures
6. Instrument and Control Procedures
7. Surveillance Tests

Radiation Protection Procedures, Emergency Plan Procedures, Security Procedures, Emergency Operating Procedures and Abnormal Operating Procedures were upgraded under separate projects.

The PUP instituted administrative controls currently in use, such as the procedure writer's guide, now documented as Standing Order G-73, *Fort Calhoun Station Writer's Guide*. The purpose of the Writer's Guide is to provide the necessary guidance for writing FCS Operating Manual procedures. The intent of the Writer's Guide is to ensure

station procedures are clear, concise, technically correct and written for the users with little or no margin for interpretation.

The PUP involved several phases. During data gathering and drafting, design bases requirements were incorporated into the draft procedures through implementation of the Writer's Guide. Information sources used during the drafting phase included, but was not limited to, the following:

1. Technical Specifications
2. USAR
3. QA Plan
4. Vendor manuals
5. OPPD and Vendor Drawings
6. Industry Standards (ANSI, ASME, IEEE, ASTM, etc.)
7. System Training Manual
8. NRC/INPO commitments
9. FCS Operating Manual
10. Experienced plant personnel
11. Configuration Changes (Mods, ECNs)
12. Design Basis Documents (When available)

After each procedure was drafted, a verification and validation was performed. The verification of procedures refers to the processes used to ensure that:

1. Procedures are technically correct,
2. Correspondence exists between the procedures and the control room/plant hardware, and
3. Language and nomenclature used in the procedures are consistent with the terms familiar to users.

The validation of procedures refers to a method for ensuring that procedures are usable by plant personnel and able to accomplish the intended objectives. Essentially, it is a simulation (or rehearsal) of the procedure, progressing through the steps in as realistic a manner as practical. After validation was performed, each procedure was sent to the Plant Review Committee (PRC) for review and recommendation of approval to the Manager-FCS. After approval and any required training, each procedure was issued for use.

Design Bases/Safety Related Operating Procedure Compliance Project

In 1990, OPPD completed the Design Bases/Safety Related Operating Procedure Compliance Project, a separate but related procedure upgrade project which satisfied the commitments associated with SEP Item 5. The objective of this project was to confirm that, concurrent with the Design Basis Reconstitution project, Safety Related Operating Procedures, Emergency Operating Procedures (EOPs), Abnormal Operating Procedures (AOPs),

and the Technical Data Book (TDB) were in compliance with the FCS Design Bases. The scope of Safety Related (SR) Operating Procedures included SR Operating Instructions and SR Operating Procedures. At the time of the project, a total of 75 OIs/OPs were classified as safety related.

Verification that the 75 OIs/OPs complied with the FCS Design Bases was accomplished by determining if the procedure complied with the requirements of the Design Basis Documents (DBDs). Because of the timing of issuance of DBDs and upgraded procedures, some upgraded procedures were verified after they had been issued. The verification included determination that the purpose or functions described in procedures were consistent with the DBDs, that setpoints or system requirements were appropriately called out in the procedures, and that there were no contradictions between functions and setpoints in the procedures and the DBDs.

The Technical Data Book was verified against applicable DBDs using a two-volume cross-reference matrix of sorted data. One volume presented TDB Figures in numerical order with the applicable DBDs listed beside them. The other matrix volume presented the DBDs in numerical order with the applicable TDB figures listed beside them. No discrepancies between the TDB and the DBD were identified during the review process.

The verification of EOP and AOP compliance with the FCS Design Bases was completed through development of EOP and AOP Technical Basis Documents. Each EOP Technical Basis Document notes the differences between the EOP and report CEN-152, Combustion Engineering Emergency Procedure Guidelines, which was used as the basis for the FCS EOPs. These differences resulted from the incorporation of plant-specific technical information and operating philosophy. The FCS-specific information was obtained from design documents. The AOP Technical Basis Documents include reference information for these procedures, including appropriate design documents. The verification and validation process for EOPs and AOPs ensured the technical adequacy of the procedures.

B.2 Current Programs and Processes

The FCS Quality Assurance Plan, Section 2.0, establishes requirements for implementing procedures which prescribe activities affecting safety. Standing Orders G-73 and G-30, *Procedures Changes and Generation*, establish responsibilities and general requirements for procedure change and generation. The Standing Orders are responsive to ANSI N18.7 - 1972, *Administrative Controls for Nuclear Power Plants*; the FCS Updated Safety Analysis Report (USAR), Section 12; and FCS Technical Specification 5.8, Procedures. The Standing Orders also incorporate commitments related to SEP Item 46, which implemented a procedures control and administration program.

Standing Order G-73 is the Writer's Guide for procedures, and is intended to provide the necessary guidance for writing of FCS Operating Manual procedures. The Writer's Guide provides consistency in matters of format, human factors considerations, and technical content, and implements the verification and validation processes.

The effectiveness of the procedures is directly dependent upon the writer's ability to obtain and apply information that exists relative to operating or maintaining the affected system or equipment. Information sources include, but are not limited to, the following:

1. Technical Specifications
2. USAR
3. Vendor Manuals
4. Industry Standards
5. Design Basis Documents
6. NRC/INPO Commitments
7. Configuration Changes

In accordance with Standing Order G-30, *Procedure Changes and Generation*, the writer is responsible for ensuring the administrative and technical accuracy of the proposed procedure/procedure change and performance of a 10 CFR 50.59 Safety Evaluation as required by NOD-QP-3, including review of any applicable Modification Request or Engineering Change Notice Safety Evaluation for consistency. The writer optionally forwards the procedure/procedure change to the Engineering Department for preparation of the Safety Evaluation if the change is specifically based on calculations or engineering analyses. This process provides reasonable assurance that procedure writers incorporate the design basis into operating, maintenance and testing procedures.

The writer then compiles a verification/validation review package. Verification and validation are defined and implemented consistent with the program initiated by the Procedures Upgrade Program (previously discussed in section B.1).

FCS Technical Specification 5.8 delineates the requirements for the review and approval of procedures that affect nuclear safety; these requirements are implemented by the Administrative Controls Standing Orders such as G-30. A procedure or procedure change is reviewed by a Qualified Reviewer (QR) who is knowledgeable in the functional area affected but is not the individual writer. The QR renders a determination in writing on whether or not cross-disciplinary review of a procedure, or change thereto, is necessary. If necessary, such review is performed by appropriate personnel.

FCS Technical Specification 5.8.2.4 requires "Qualified Reviewers meet or exceed the respective qualifications for either Supervisors Requiring an AEC License, Professional-Technical Personnel, or Technical Support Personnel, as specified in ANSI N18.1 - 1971. Personnel recommended to be QRs shall be reviewed by the PRC and approved and designated as such by the PRC Chairman."

A procedure, or change thereto, is reviewed by the Department Head designated by Administrative Control Standing Orders as the responsible Department Head for that procedure. The review includes a determination of whether or not a 10 CFR 50.59 safety evaluation is required. If a 10 CFR 50.59 safety evaluation is not required, the

procedure, or change thereto, is approved by the responsible Department Head or the Manager-FCS prior to implementation.

If the responsible Department Head determines that a procedure, or change thereto, requires a 10 CFR 50.59 safety evaluation, the Department Head renders a determination in writing of whether or not the procedure, or change thereto, involves an Unreviewed Safety Question (USQ) and forwards the procedure, or change thereto, with the associated safety evaluation to the Plant Review Committee (PRC) for review in accordance with Technical Specification 5.5.1.6.a. If a USQ is involved, NRC approval is required prior to implementation of the procedure, or change. The PRC recommends approval or disapproval to the Manager - FCS.

Procedure changes requiring training are reviewed and approved, with issuance held until the requirements of the Procedure Training Request are met. Completion of training before or after procedure approval may be specified. After review and prior to implementation, the responsible Department Head within the functional area or the Manager-FCS must approve the procedure or change in accordance with Standing Order G-30.

B.3 Assessments, Surveillances and NRC Inspections

A review of Quality Assurance Surveillances (1991 to present) related to station procedure control did not find any programmatic deficiencies in the procedures, their implementation or results. Since January 1995, the FCS Quality Assurance Department has ensured the acceptability of FCS procedures by implementing the Biennial Procedure Review Oversight Program. Quality Assurance Manual Procedure #14 (QAM-14) is used to administer the program. The sampling plan used in QAM-14 provides a 95% confidence level that there are no more than 5% nonconforming procedures in the total procedure population being reviewed. To date, the FCS procedures have been in compliance with the criteria specified in QAM-14. This indicates the overall effectiveness of the procedure improvements and current process. Significant discrepancies or deficiencies found in individual FCS procedures are identified and resolved through the corrective action program.

In Licensee Event Report (LER) 91-08, OPPD voluntarily reported the discovery of inappropriate Technical Specification (TS) surveillance requirements for Reactor Protective System bistables, and inadequate documented verification of the incorporation of all applicable TS surveillance requirements into FCS surveillance test procedures. The corrective actions included implementation of a TS Verification Action Plan. Surveillance procedures were compared to the TS to ensure each required surveillance has a corresponding surveillance procedure which meets the intent of the TS. To ensure that the information gathered during this phase of the project was captured, a surveillance test/technical specification matrix was developed and installed in an on-line, main frame computer system that can be readily accessed. This was completed in November 1992.

However, a surveillance test to monitor Boric Acid Storage Tank level was not properly reviewed during implementation of the Action Plan due to administrative oversight. This procedure did not satisfy TS requirements, a condition reported by LER 93-01.

In March 1993, the FCS staff began a detailed effort to ensure that the content of individual station surveillance tests met the intent of the requirements of the technical specifications. A review of all surveillance tests assigned to system engineers was performed by the respective system engineer. In addition, an independent review was performed of approximately 10% of all the surveillance tests.

The results of these reviews uncovered a number of minor discrepancies, none of which were contrary to the requirements of the technical specifications, but rather enhancements to the test program, individual surveillance tests, or enhancement to the technical specifications. An action plan that included each discrepancy was developed with specific assignment to station staff, by primarily to system engineering. Currently, the action plan is scheduled to be complete in April 1997.

NRC Inspection Report 91-01, Electrical Distribution System Functional Inspection, concluded that the FCS DBDs were generally comprehensive, user friendly and provided a good information source for the engineering staff. Design information was noted as being readily retrievable and accurate. The procedures developed through the FCS procedure rewrite program were noted as appearing greatly improved over earlier revisions.

NRC Inspection Report 94-01 documented a station blackout team inspection. The report states that the team found the calculations and analyses to be technically sound and in accordance with approved guidelines (i.e., the design basis). The team also determined that the station blackout coping procedures adequately addressed control room and local operator actions necessary to establish and maintain decay heat removal, minimize RCS inventory loss, minimize DC loads and restore the availability of vital components.

NRC Inspection Report 96-01, a Resident Inspector report, included a review of activities relative to plant practices, procedures, and parameters detailed in the USAR (a portion of the plant's design basis). The inspectors verified that the USAR wording was consistent with the observed plant practices, procedures, and parameters. NRC Inspection Report 96-03, another Resident Inspector report, stated that in the Maintenance area good procedural compliance was noted.

B.4 Conclusion

FCS procedures were rewritten/upgraded in the late 1980s through the early 1990s. Technical content and accuracy of the procedures continues to be verified including verification of correspondence between the procedures and plant hardware. Responses to Requests C and E provide OPPD's rationale for concluding that the plant hardware is

consistent with the design basis. The verification process and the correspondence between the procedures and plant hardware provide OPPD with reasonable assurance that the FCS design bases are reflected in the operating, maintenance and testing procedures used since the implementation of these programs, projects, and processes.

Omaha Public Power District Response to NRC Request C

(C) Rationale for concluding that system, structure, and component configuration and performance are consistent with the design bases

Executive Summary

Omaha Public Power District (OPPD) has reasonable assurance that the configuration and performance of the Fort Calhoun Station (FCS) systems, structures, and components are consistent with the design bases. This reasonable assurance is based on the following items:

1. Reconstitution of the FCS design bases (Section C.1)
2. The process used to disposition missing or conflicting design documents (Section C.2)
3. Verification that the reconstituted design bases and design documents are accurate and complete (Section C.3)
4. Verification that the FCS physical configuration is consistent with the reconstituted design basis (Section C.3)
5. The system oriented assessments conducted by OPPD, including audits based on Safety System Functional Inspections (Section C.3)
6. Inspections by the NRC of the reconstitution of the design basis and the processes, programs, and procedures for changing and controlling the configuration of the station (Section C.4)
7. Verification that the USAR is consistent with the design bases (Section C.5)
8. Effective control of design changes made after the design bases reconstitution which ensures the plant continues to conform to the design basis (Section C.6)
9. Additional verification of the USAR and design bases currently being conducted by OPPD using Nuclear Energy Institute guidance (Section C.7)

Background

Following a Safety System Outage Modification Inspection (SSOMI) by the NRC in December 1985, OPPD agreed to organize the original Fort Calhoun Station (FCS) design basis information in a more manageable form to ensure that the original design margins are not unintentionally abrogated. The design bases were documented and reconstituted by

the Design Basis Reconstitution Project. Various aspects of this project were tracked by Safety Enhancement Program Items 4, 5, 6, and 7.

C.1 Design Basis Reconstitution

The FCS design bases were reconstituted by the Design Basis Reconstitution Project which began in March 1987 and completed the development of the Design Basis Documents (DBDs) in April 1990. OPPD contracted with Combustion Engineering and Stone & Webster Engineering for the production of the DBDs.

One of the challenges to the Design Basis Reconstitution Project was the recovery of the reference design data necessary to develop the design basis documents. Because of the vintage of the plant, current record keeping and storage requirements were not in effect at the time of original plant design. Therefore, the original design and construction records were not always properly stored, maintained, indexed or even identified as being required. OPPD also contacted the architect/engineer, reactor vendor, and major equipment suppliers for FCS design records they possessed. These searches yielded thousands of records containing design basis information.

The Design Basis Reconstitution Project developed 47 Design Basis Documents (DBDs). Of these documents, 33 were specific to selected systems and 14 were topical documents covering a subject that spanned many of the DBDs. The goal of the project was to develop system design basis documents which would reflect the current condition of the plant. Selection of systems needing DBDs was dependent on the safety significance of the system, frequency of modification, complexity of the system and importance of the system for sustained safe plant operation. The following is a list of the System level Design Basis Documents (SDBDs) that were developed for FCS.

SDBD-AC-CCW-100	Component Cooling Water
SDBD-AC-RW-101	Raw Water
SDBD-AC-SFP-102	Spent Fuel Pool Cooling
SDBD-CA-IA-105	Instrument Air
SDBD-CH-108	Chemical & Volume Control
SDBD-CN-110	Plant Communications
SDBD-DG-112	Emergency Diesel Generators
SDBD-DW-113	Demineralized Water
SDBD-FP-115	Fire Protection
SDBD-FW-116	Feedwater
SDBD-FW-AFW-117	Auxiliary Feedwater
SDBD-HG-122	Nitrogen & Hydrogen Gas
SDBD-MS-125	Main Steam
SDBD-RC-128	Reactor Coolant
SDBD-SI-130	Shutdown Cooling
SDBD-SI-CS-131	Containment Spray
SDBD-SI-HP-132	High Pressure Safety Injection

SDBD-SI-LP-133	Low Pressure Safety Injection
SDBD-SL-PAS-134	Post Accident Sampling
SDBD-SL-135	Primary and Secondary Sampling
SDBD-VA-AUX-138	Auxiliary Building HVAC
SDBD-VA-CON-139	Containment HVAC
SDBD-VA-CR-140	Control Room Habitability
SDBD-WD-144	Waste Disposal
SDBD-EE-200	120 VAC Vital Distribution
SDBD-EE-201	AC Electrical Distribution
SDBD-EE-202	DC Electrical Distribution
SDBD-EE-203	Cathodic Protection
SDBD-COMP-300	ERF Computer & QSPDS Computer
SDBD-CONT-501	Containment
SDBD-AUX-502	Auxiliary Building
SDBD-STRUC-503	Intake Building
SDBD-STRUC-504	Security Building

Plant level design basis documents were developed when a specific topic covered more than one system. The following is a list of the Plant Level Design Basis Documents (PLDBDs) for FCS.

PLDBD-ME-10	Pipe Stress & Supports
PLDBD-ME-11	Internal Missiles & HELB
PLDBD-EE-21	Electrical Equipment
PLDBD-IC-30	Instrumentation Installation
PLDBD-IC-32	Instrumentation & Controls Systems
PLDBD-CS-50	External Missiles
PLDBD-CS-51	Seismic Criteria
PLDBD-CS-52	Heavy Loads
PLDBD-CS-54	Geotechnical
PLDBD-CS-55	Masonry Walls
PLDBD-NU-61	Regulations, Codes and Standards
PLDBD-NU-63	Personnel Protection
PLDBD-EV-70	Site Meteorology

Not all systems at FCS were candidates for design basis documents based on the selection criteria above. The following is a listing of those systems.

Auxiliary Steam System
Chemical Feed System
Circulating Water System
Lubricating Oil System
Potable Water System
Sanitary & Storm Drains
Toxic Gas Monitoring System

Turbine Supervisory System
Heater Vents & Drains
Vacuum Priming System

To ensure conformance with the design basis for modifications designed and installed prior to the Design Basis Reconstitution, OPPD conducted a safety evaluation check of Modification Design Change packages (for safety related and nonsafety related modifications that had potential to impact safety systems) that were installed since initial commercial operation. Concerns identified through this review were dispositioned using the DBD open item process (described in Section C.2).

To ensure that activities such as modifications, procedure changes and safety evaluations could be conducted while the DBR Project was in progress, controls were established to ensure that design basis margins were not abrogated. These controls provided:

1. An interim position and guidance for selection of design inputs and re-creation of design bases on an as needed basis, and

2. A review of modifications by third parties, including OPPD Quality Assurance and outside Architect/Engineer firms.

Prior to starting the development of the design basis documents, writer's guides were written, reviewed and approved by OPPD staff. In order to maintain consistency between the documents, writer's guides for both the SDBDs and PLDBDs were created. During the development of the DBDs, it was important to clearly define the scope of each design basis document. Without a clear definition of the scope, system boundaries or topics may have overlapped. To ensure that an overlap did not occur, system boundary drawings were developed first. These boundary drawings, similar to Piping & Instrumentation Diagrams (P&IDs) containing major components, aided the DBD preparer in clearly understanding the scope of the design basis documents during development.

The initial step in the preparation of each system DBD was the development of the requirements section. The primary references for the requirements sections were:

1. The USAR (and FSAR),
2. The Commitment Tracking System,
3. System modification packages, and
4. NRC Safety Evaluation Reports.

The following is a listing of additional resources used to assist in the development of the DBDs:

1. Technical Specifications
2. Equipment Specifications

3. Calculations
4. Test Reports
5. Correspondence
6. Licensing Correspondence
7. Contract Documents
8. Drawings
9. Vendor Information
10. Design Changes (Modifications Packages)

C.2 Missing Documentation

During the process of reconstituting the design bases for FCS, information or references were not always available to fully define or support the design requirements or to verify the adequacy of the design bases. In many instances, because of the vintage of the plant, that documentation did not exist. There also were instances where controlled documentation contained conflicting data. In order to identify, evaluate and classify the safety significance of these discrepancies, OPPD initiated a DBD Open Item Program. These open items ranged from minor discrepancies having no impact on the safety of the plant to issues potentially requiring notification of the NRC. When a discrepancy was found in the design basis, the initiator identified it with an open item number. OPPD procedure PED-QP-29, *Evaluating, Reconstituting, and Closing Design Basis Document Open Items*, was used to properly evaluate and classify the open items and prioritize them based on their safety significance. Each open item was assigned a category ranging from Category 1 (potentially reportable) to Category 6 (least serious).

Category 1 Open items which identify specific design deficiencies that can adversely affect safety-related systems, structures, components or equipment and are potentially reportable to the NRC

Category 2 Open items which (1) address safety-related equipment or components that perform an active function in satisfying plant system operations requirements as specified in the Technical Specifications and do not meet the reportability criteria of 10CFR50.72, 10CFR50.73 or 10CFR21; or (2) identify conflicting requirements or missing important documentation, or (3) identify missing 10CFR50.59 evaluations.

Category 3 Open Items which (1) address safety-related equipment or components that do not perform an active function in satisfying plant operational requirements as specified in the Technical Specifications and do not meet the reportability criteria of 10CFR50.72, 10CFR50.73 or 10CFR50.21; or (2) identify missing documentation that is needed to confirm compliance with structural, seismic, pipe stress, pipe support, or HELB design requirements; (3) identify missing supporting documentation for 10CFR50.59 evaluations, or (4) identify unresolved code issues or missing

calculations/analysis required to prove conformance with regulatory commitments.

Category 4 Open Items which are being resolved on a generic basis (i.e., NRC's unresolved issues, or other issues identified by industry groups).

Category 5 Open Items which reflect minor discrepancies. They have no safety significance and are not necessary to improve the overall understanding of the design basis.

Category 6 Open Items for which the appropriate paperwork has been initiated to closeout/resolve the condition.

During the course of reconstituting the design bases for FCS, a total of 1715 open items were identified, evaluated, classified and dispositioned. A list of the category subtotals follows.

Category 1: 10
Category 2: 191
Category 3: 839
Category 4: 13
Category 5: 586
Category 6: 76

Because the Category 1 open items were potentially reportable, they were dispositioned immediately after identification. The Category 2 open items were closed approximately twelve months (April 1991) after the completion of the reconstitution project. The remaining open items were evaluated and determined to have minimal impact on the quality of the DBDs and have been scheduled for closure based on resource availability. As of February 4, 1997, a total of 200 open items remain in the following categories.

Category 1: 0
Category 2: 0
Category 3: 93
Category 4: 0
Category 5: 101
Category 6: 6

These open items are not safety significant and are being actively pursued; completion of closure is currently scheduled for June 1997.

C.3 Design Basis Verification

During reconstitution of the design basis for FCS, OPPD began to verify that the design documents were in agreement with the configuration of the plant. Verification during the DBD project was accomplished by physical walkdowns, procedure reviews, licensing commitment reviews, functional checks, safety evaluation checks, and system oriented assessments.

Walkdowns:

Over the years, OPPD has conducted various walkdowns such as the Piping and Instrumentation Diagram (P&ID) drawing walkdown, Drawing Update Project walkdown, Computerized History and Maintenance Program (CHAMPS) walkdown, NRC Bulletins 79-02 and 79-14 walkdowns, and the Environmental Qualification Program walkdown. These walkdowns served to document the as-built condition of the plant, which provided assurance during the Design Basis Reconstitution Project that the physical plant was correctly reflected in the DBDs. The results of these walkdowns were used to verify that the contents of the DBDs reflect the as-built configuration and regulatory requirements.

Procedure Reviews:

The Operating Procedures, Emergency Operating Procedures, Abnormal Operating Procedures and Technical Data Book were verified to be in compliance with the design bases as part of the Design Bases/Safety Related Operating Procedure Compliance Project described in the response to Request B.

Functional Checks:

This verification confirmed that safety systems and other selected systems would perform their intended functions. A review of the FCS operating experience was done to verify that systems performed their normal operating functions as designed.

Systems required to operate in the post-accident mode were reviewed to verify that the ability to perform their intended functions was not compromised as a result of the modification process. This verification included reviews of post modification functional testing, reviews of the surveillance tests, and review of operating experience.

Safety Evaluation Checks:

Concurrent with the reconstitution effort, OPPD conducted a review of documentation packages for safety related modifications (and non-safety related modifications which had a potential to impact safety systems) which were installed since commercial operation in 1973. There were 1304 modifications reviewed for potential impact on safety related systems. Of these, 1001 were identified for review of Unreviewed Safety

Question (USQ) Evaluations. The 10 CFR 50.59 evaluations associated with these modifications were reviewed to insure that no unreviewed safety questions were overlooked. Fifty-nine modifications were dispositioned through on-going OPPD programs. One hundred and ninety-seven modifications lacked sufficient documentation to adequately determine whether a USQ existed and were dependent on the closure of pre-identified open items. Closure of these open items resulted in no additional identification of modifications with Unreviewed Safety Questions. Based on the project's review, 16 modifications were determined to contain USQs. They were assigned to a Category 1 open item and immediately resolved in accordance with PED-QP-29, *Evaluating, Reconstitution, and Closing Design Basis Document Open Items*.

System Oriented Assessments:

Since 1987, OPPD has completed 9 (including one follow-up) audits based on NRC guidance for Safety System Functional Inspections (SSFIs). During these audits, vertical slice evaluations were performed on the following 8 systems: Auxiliary Feedwater, Instrument Air, 120 VAC Vital Distribution, Component Cooling Water, 125 VDC, Raw Water, Stator Cooling, and Diesel Generator. The purpose of these audits was to verify the operational readiness of the systems (including system modifications, testing, maintenance and operation) to perform required safety functions in conformance with the current licensing bases. This verification also served to validate the effectiveness of the Design Basis Reconstitution Program.

The follow-up SSFI audit was performed in 1991 to ensure that corrective actions were performed for deficiencies identified during the Instrument Air, Component Cooling Water, and Auxiliary Feedwater SSFI audits. This follow-up audit also included verifications that deficiencies identified by OPPD prior to the NRC Electrical Distribution System Functional Inspection (EDSFI) were corrected or on track for correction. During this audit, more than seventy corrective action documents were verified to have corrective actions taken. Corrective actions for the identified deficiencies included revisions to procedures, drawings, and various maintenance activities; establishment of the interim set-point control program; equipment label changes; and modification and design evaluations.

The subsequent SSFI audits each generated approximately three concerns. These audits show a continuing trend of improved conformance of the plant to its design bases. The deficiencies dealt with items such as inconsistencies between USAR values and calculations, inadequate post-maintenance testing of equipment, and administrative deficiencies with Design Basis Documents, plant procedures, and drawings.

Review of corrective action documents generated as a result of these inspections noted that corrective actions such as updates to procedures, System Training Manuals and drawings, changes to operation and testing procedures, and corrections to the Computerized History and Maintenance Program System (CHAMPS) equipment database have taken place to correct the identified deficiencies.

During the months of October and November 1994, OPPD personnel conducted the Service Water System Operations Performance Inspection (SWSOPI). This assessment determined that the Raw Water, Component Cooling Water and related supporting systems are in good condition. The assessment also concluded that the systems are operated within the design criteria for FCS.

C.4 NRC Inspections

In addition to assessments conducted by OPPD which evaluated effectiveness of the FCS Design Basis Reconstitution Project and configuration control process, the NRC has conducted several major inspections of OPPD's programs. In April 1990, the NRC conducted an Electrical Distribution System Functional Inspection (EDSFI), as documented by Inspection Report 91-01, dated May 20, 1991. The inspection team reviewed selected modifications, design calculations, and design basis documents. Additionally, the inspection team conducted interviews and reviewed training records, corrective action records, maintenance records and surveillance test records. The review focused on the off-site power supply grid and associated on-site, safety-related busses and loads.

During the EDSFI, the team noted a number of strengths and relatively few weaknesses. The team noted that the design basis documents were generally comprehensive, user friendly, and provided a good information source to the engineering staff. Design information was readily retrievable and accurate.

An NRC/NRR Special Audit of Control Processes for Commitments and Current Licensing Basis was documented in a report dated October 19, 1993. The staff conducted audits at a cross section of reactor plants to assess the processes used by licensees for controlling commitments that affect the plants' current licensing basis, including maintaining and updating the final safety analysis report. At FCS, the audit team found that commitments which affected the USAR were captured by the USAR update process. The team's review of several plant modifications to identify changes to plant systems and verify incorporation of the changes in the USAR showed that all affected text descriptions and system drawings in the USAR were properly revised to reflect the associated modifications.

C.5 USAR Verification Project

The NRC SSOMI report and Inspection Report 91-19 also identified problems with the adequacy of information in the USAR. Specifically, certain design information and safety analyses sent to the NRC were not incorporated, and a radiation monitor was still described years after its removal from service. As corrective action, OPPD implemented the USAR Verification Project in January 1993 to accomplish 3 objectives:

1. Verification of the accuracy of information contained in the USAR utilizing information (design basis documentation) gathered during the Design Basis Reconstitution Project

2. Verification that a proper review and inclusion of information required by 10 CFR 50.71(e) had been completed

3. Assurance that processes for USAR update and review of procedures were adequate for maintaining information current and accurate

The scope of the USAR Verification Project was to review and verify the information in Sections 1-11 and 14. The remaining sections (12 and 13) and the appendices (A through N) were outside the scope of the Project because they were either archived, relocated out of the USAR, or had recently been verified by other means. Section 13 was archived. Appendix A is updated annually in accordance with the QA Plan. Appendices B through D and G through L were archived. Appendix E was incorporated into USAR Section 12, Conduct of Operations. Appendix F was updated by Design Engineering. Appendix M was updated under the normal USAR update process. Appendix N was updated in 1992.

The USAR review and update procedure (Quality Procedure NOD-QP-16, *Updated Safety Analysis Report*) was reviewed and appropriately upgraded to ensure the USAR is maintained accurate and current. A USAR Writer's Guide was developed to provide guidelines in matters regarding content, format, style and detail. The Writer's Guide is an appendix to NOD-QP-16.

Verification was accomplished by utilizing the information gathered during the Design Basis Reconstitution Project. NRC Safety Evaluation Reports (SERs), NRC Generic Letters (GLs) and NRC Inspection and Enforcement Bulletins (IEBs) issued from 1973 through 1994 were reviewed. If inaccuracies or omissions were discovered during the verification process, corrections and additions were initiated along with a justification for each. Each verified section, with the applicable changes/justifications, was sent to System Engineering, Design Engineering and Operations for an independent review. The System Engineer, Design Engineer and the Operations Engineer not only reviewed the changes for accuracy and concurrence, but reviewed the entire section for accuracy. Any necessary additional changes, along with justifications, were incorporated before review and approval by the Plant Review Committee and subsequent incorporation into the next scheduled USAR update.

The USAR Verification Project was completed in July 1995. Recently completed and ongoing self assessments (described in Section C.7 below) have identified insignificant discrepancies between the USAR and the plant configuration. These discrepancies have been entered into the Corrective Action Program for resolution.

NRC Inspection Reports 96-01 through 96-04 and 96-08 verified that, in the areas inspected, the USAR wording was consistent with the observed plant practices, procedures, and/or parameters.

C.6 Engineering Design and Configuration Control

Engineering Design and Configuration Control processes have been upgraded to ensure that the FCS design bases are available and maintained during design change development and installation of new designs. The Configuration Control process delineates responsibilities, interfaces, review, approval, and training requirements. The control processes ensure that only approved configuration changes are made in the station, and engineering documents, and databases are accurately updated in a timely manner. Clear guidance is provided on the various types of authorized configuration changes. Procedure PED-QP-2, *Configuration Control*, assists the Design Engineer in determining the type of configuration change to use. These processes are discussed in more detail in the response to Request A.

C.7 Ongoing Verification Activities

In order to provide additional assurance of conformance with the design bases, OPPD is currently completing assessments of selected FCS systems using the Nuclear Energy Institute (NEI) *Guidelines for Assessing Programs for Maintaining the Licensing Basis* (NEI 96-05). Deficiencies found are entered into the Condition Report system for resolution. The assessment of the Chemical and Volume Control System is complete, and assessments of the Safety Injection and Instrument Air systems are ongoing. OPPD will complete assessments of all remaining FCS safety related and safety significant systems no later than February 1, 1999.

C.8 Design Basis Document Maintenance

In order to maintain the dynamics of the Design Basis Reconstitution Program, OPPD institutionalized the Design Basis program by conducting training for the users and by issuing procedures that control the changes to the documents. OPPD procedure PED-GEI-3, *Preparation of Design Change Packages*, requires that the design engineer use the DBDs as a source for the design input of the design change package. Any changes in the design basis due to the modification result in revision to the appropriate DBDs. The actual update of the design basis documents is controlled by PED-QP-13, *Design Basis Document Control*. This procedure establishes the requirements and responsibilities for control and approval of revisions to the DBDs. When proposed changes are submitted by other than the design basis document Sponsor an additional technical review is required per procedure GEI-12, *Design Engineering Review of Design Basis Document Revisions*. This procedure insures that the proposed configuration change is technically correct, substantiated by design documentation and is evaluated for impact on other Fort Calhoun Station documents.

C.9 NUMARC 90-12

The OPPD Design Basis Reconstitution Program, although completed before the issuance of NUMARC 90-12, contained all the attributes necessary for a successful program as outlined in NUMARC 90-12. This document was assembled by a team representing 12 utilities, 4 NSSS vendors, and 4 A/E firms. Input to the guideline was also received from INPO based on trips to utilities that had mature programs in place. Some of the lessons learned and reflected in NUMARC 90-12 are the result of evaluation of the FCS program.

C.10 Conclusion

The design bases reconstitution, verification, maintenance, and assessment activities described above provide OPPD with reasonable assurance that the configuration and performance of the FCS systems, structures and components are consistent with the design bases. Discrepancies identified during these activities have been and will continue to be resolved through appropriate corrective action processes.

Omaha Public Power District Response to NRC Request D

(D) Processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, action to prevent recurrence, and reporting to NRC.

Executive Summary

The Corrective Action Program (CAP) is used for the identification, reporting, documenting, controlling, tracking and trending of conditions adverse to quality at Fort Calhoun Station (FCS). This applies to conditions that represent a failure to meet requirements or expectations, such as malfunctions, deficiencies, defects, deviations, non-conformances, or abnormal occurrences. The Corrective Action Program provides reasonable assurance that problems are identified, required reports are made to the NRC, and effective corrective actions are implemented, all on a schedule commensurate with the nuclear safety significance. This assurance is based on the effective implementation of the processes making up the Corrective Action Program, summarized as follows:

1. The Condition Reporting (CR) system, proceduralized in Standing Order R-2, implements the Corrective Action Program and provides the primary tool for identifying, categorizing, tracking and documenting correction of conditions adverse to quality. (Section D.2)

2. Standing Order R-2 requires personnel to identify conditions adverse to quality including nonconformances in design basis documents, as-built plant and procedures. (Section D.2)

3. Design concerns are formally documented in the CR system and evaluated by both the Corrective Action Group and the Condition Review Group. (Section D.2 and D.5)

4. Identified conditions adverse to quality are evaluated to determine their impact, using the Operability Determination Program which provides a means of formal documentation for operability evaluations. (Section D.3)

5. The Condition Review Group evaluates identified conditions adverse to quality and assigns a priority for timely corrective action based on their significance to safety. (Section D.5)

6. The CR system is used to track identified conditions adverse to quality through completion of corrective actions and is also used to trend CRs to identify potentially significant adverse conditions. (Sections D.5, D.7 and D.8)

7. The Root Cause Analysis Program provides a formal method for root cause determination of significant conditions adverse to quality and the identification

and implementation of actions to prevent recurrence and correct any generic implications. (Section D.6)

8. If it is necessary to operate FCS with a degraded condition that does not conform with the design basis, the condition is formally analyzed and documented using the Safety Analysis for Operability process, which includes the performance of a 10 CFR 50.59 safety evaluation. (Section D.3)

9. The Reportability Determination Program provides a method to ensure that identified conditions adverse to quality are evaluated for reportability to the NRC under requirements including 10 CFR 50.72, 10 CFR 50.73, 10 CFR 50.9, 10 CFR Part 21, and the Technical Specifications. (Section D.4)

10. Audits, surveillances, and assessments of the corrective action program and its implementation have been conducted by the Quality Assurance Department, the Nuclear Safety Review Group, and independent third parties. These audits, surveillances, and assessments have identified no significant deficiencies in the procedures, including implementation or results. However, an ongoing problem with timeliness of responses and deficiency correction has been noted, and is being addressed through increased management attention. (Section D.9)

11. Inspections by the NRC of the current FCS Corrective Action Program have identified no significant or programmatic deficiencies in the procedures, including implementation or results. (Section D.9)

D.1 Corrective Action Program Prior to the Condition Report System

Prior to implementation of the current Condition Report (CR) system at FCS, several systems comprised the Corrective Action Program. The Operational Non-conformance (ONC), Incident Report (IR), and Corrective Action Report (CAR), systems were the Corrective Action Programs in general use up to September 1995. In addition, the Design Basis Reconstitution program had its own Open Item identification and tracking process.

Operational Non-conformance (ONC) System

The ONC System was used to identify and document resolution of nonconforming conditions found in any "operable" system, structure, component or software identified during plant operation, testing, inspection or maintenance. A nonconforming condition is defined as a deficiency or noncompliance in characteristic, documentation or procedure which renders the quality of an item unacceptable or indeterminate. The ONC was a separate screening process that had been operated by Quality Assurance. The ONC system was principally used by Engineering to disposition material conditions adverse to quality in operating equipment.

Incident Report (IR) System

The IR system was the primary mechanism used to report operating anomalies and deficiencies affecting the design basis, licensing basis, or configuration of FCS. Typically, when an employee identified an item of concern, it was discussed with the employee's supervisor before being considered for further reporting. When initiated, an IR was reviewed by the Shift Technical Advisor (STA) and Shift Supervisor for reportability, and an operability determination was made for equipment governed by Technical Specification Limiting Conditions for Operations. The STA would also perform a safety assessment for any event that placed the plant in an abnormal situation or any plant parameter that affected or reflected an abnormal indication of a safety related system.

The IR was then reviewed at the weekly Incident Evaluation Team (IET) meeting. The IET was established as a multidisciplinary subcommittee of the Plant Review Committee (PRC). For each IR, the IET:

1. Ensured reportability requirements were met,
2. Determined significance,
3. Assigned appropriate responsibility, depth of investigation, and documentation requirements,
4. Reviewed for operational nonconformances, and
5. Identified relevant industry experience.

As another multidisciplinary management review, the PRC then reviewed each IR with the assignments, recommendations and comments made by the IET and modified them as applicable.

The IR assignee was responsible for initiating, completing, and documenting corrective actions to ensure issues related to the IR were resolved. One of the actions may have been to perform the ONC screening and provide a basis for correcting, removing, or accepting the condition. ONCs were incorporated as part of the IR process which allowed ONCs to be approved, closed out and trended through the IR system.

The IR system was replaced with the CR system in September of 1995. The IRs with open corrective actions were left on the IR system for completion. Upon completion of corrective actions, all IRs are reviewed by an IR coordinator to ensure all issues are resolved and documented. If the IR is considered significant, the IET and PRC will review the IR for closure, again ensuring both a multidisciplinary and managerial perspective of the completed corrective actions.

Corrective Action Report (CAR) System

The other system in common use was the Corrective Action Report (CAR). The CAR system was used primarily by the Quality Assurance organization to record conditions adverse

to quality identified during their audit process; however, it was available for use by all personnel. The corrective actions were reviewed and accepted by QA management or for a significant deficiency, by the Safety Audit and Review Committee (SARC) Chairperson. The deficiencies identified by the CAR system tended to be more programmatic in nature.

Design Basis Reconstitution Open Item Tracking

During the process of reconstituting the design bases for FCS, information or references were not always available to fully define or support the design requirements or to verify the adequacy of the design bases. In order to identify, evaluate and classify the safety significance of these discrepancies, OPPD initiated a DBD Open Item Program. (This program is also discussed in the response to Request C.) These open items ranged from minor discrepancies having no impact on the safety of the plant to issues potentially requiring notification to the NRC. When a discrepancy was found in the design basis, the initiator identified it with an open item number. These open items were not entered into the corrective action process, but they were tracked on a separate open action item list. OPPD procedure PED-QP-29, *Evaluating, Reconstituting, and Closing Design Basis Document Open Items*, was used to properly evaluate and classify the open items and prioritize them based on their safety significance. Each open item was assigned a category ranging from Category 1 (potentially reportable) to Category 6 (least serious). If an open item was found to be significant to plant safety or reportable to the NRC, the deficiency was then converted into the applicable corrective action process for resolution and tracking of corrective actions.

D.2 Transition/Upgrade - Condition Report System

The corrective action programs under the Incident Report (IR), Corrective Action Report (CAR), and Operational Non-conformance (ONC) systems were considered sound but inefficient. There was also some confusion among personnel as to which system should be used to report a problem. Therefore, these processes were benchmarked by interviews, procedure reviews and direct observations. The corrective action processes at the leading industry performers were also reviewed. This information was assessed and the decision was made to develop an improved integrated corrective action program.

The new corrective action system, entitled the Condition Report (CR) system, was implemented in September 1995. Plant personnel were trained on the new system. The CR system is procedurally controlled by Standing Order R-2, *Condition Reporting and Corrective Action*. The CR system was developed to provide a consistent approach to effect actions and management oversight, to properly characterize and prioritize problems, and to provide for permanent fixes to problems. Benefits in combining these systems were consistency in threshold, significance, status, trending, and closure of issues important to safe, efficient, and reliable operation.

The Corrective Action Group (CAG) was established to provide an integrated administration of the Corrective Action Program at FCS. The CAG includes experienced personnel familiar with plant configuration, operation, and administrative controls; some of these personnel have held Senior Reactor Operator licenses and have served as Shift Supervisors. As discussed later in Section D.5, the CAG presents preliminary and background information on each CR to the management Condition Review Group. The CAG also includes a supervisor responsible for coordinating and administrating the Root Cause Analysis process. The CAG personnel can provide expert assistance to anyone using the CR system.

Standing Order R-2 requires personnel to use the CR system to identify conditions adverse to quality, including nonconformances in design basis documents, as-built discrepancies in plant systems, and procedure deficiencies. The CR system is electronically available to everyone, on a Local Area Network; however, a paper form or CAG assistance is also available for initiation of CRs. In the event an individual discovers a condition requiring immediate response by Operations (e.g., fire, flooding, medical emergency, possible equipment failures, etc.) the individual immediately contacts the Control Room. For events that require immediate response by Security (e.g., tampering, assault, bomb threats, unauthorized individuals, etc.) the individual will immediately contact the Central Alarm Station (CAS). The CR will be generated after the nonconforming condition or event has been stabilized.

The CR process was designed to encourage timely problem identification at all levels of the organization. No management approval is required for any employee to initiate a CR on any concern or deficiency. A low threshold of reporting is maintained, with the philosophy being "**when in doubt, report it**." The expectation that conditions be immediately reported is supported by senior management, as demonstrated by the Vice President's request for all supervisors to make the expectation clear to their personnel. This expectation was included in General Employee Training which is required for all personnel having unescorted access to the protected area. Experience to date shows that this expectation is being met.

D.3 Operability Determination Program

Formal operability evaluations at FCS are controlled by Nuclear Operations Division procedure NOD-QP-31, *Operability and Reportability Determinations*. The NOD-QP-31 process has been designed to implement guidance on operability determinations provided in NRC Generic Letter 91-18 and Inspection Manual Part 9900. NOD-QP-31 applies to all non-routine events and conditions discovered during activities associated with the operation of FCS. Such activities may include plant operations, maintenance, testing, analysis, surveillance or any other process which reveals the potential inoperability of specified structures, systems, or components (SSCs).

After initiating a CR, the originator ensures notification of the Shift Technical Advisor and/or the Shift Supervisor. The Shift Supervisor is responsible for completion

of an initial operability and reportability determination. Initial operability determinations are made as soon as reasonably possible following discovery of a deficiency. Determinations are generally made within 24 hours, even though complete information may not be available. Initial determinations are revised appropriately as new or additional information becomes available. Most initial determinations are performed without completion of NOD-QP-31 documentation.

For some conditions, the SSC is clearly inoperable. If the affected SSC is considered operable based on available information but further evaluation is required, the Shift Supervisor contacts appropriate management personnel to initiate a documented NOD-QP-31 operability evaluation.

Positive determinations of operability must be justified and include, as applicable, a technical discussion of why the concern identified does not prevent the item from fulfilling its intended safety function(s). This demonstrates that the item is not exceeding its design basis specified in reference documents, or the item maintains the ability to provide minimum parameter values (e.g., flow, pressure, voltage), and does not create any inter-system effects. The discussion addresses issues such as common mode failure, seismic event, station blackout, and other design basis events.

Following preparation and review of an NOD-QP-31 operability evaluation, it is forwarded to an appropriate level of management for concurrence. Following this concurrence, the Shift Supervisor takes appropriate actions based on the results of the operability determination. The Plant Review Committee (PRC) also reviews these evaluations.

Procedure NOD-QP-22, *Preparation and Approval of a Safety Analysis for Operability (SAO)*, defines a process at FCS similar to the industry Justification for Continued Operation (JCO). An SAO provides technical justification for continued plant operation when unexpected events or conditions arise that are contrary to the established design basis for FCS. In the event that a component is determined to be inoperable, and there is reasonable assurance of safety, operation may be acceptable during the corrective action phase of the problem. An SAO package includes documentation of the issue, including an evaluation in accordance with 10 CFR 50.59 requirements, and documentation of necessary compensatory and corrective actions. Following appropriate management review, an SAO is reviewed by the PRC and approved by the PRC Chairman.

D.4 Reportability Determination Program

The Condition Reporting system is used to initiate and coordinate the reportability process for FCS. Several procedures govern portions of the process. Assigned personnel provide support as required for research, analysis, and review associated with reportability determinations.

The Shift Supervisor is responsible for completing and making prompt verbal reportability determinations. The Shift Technical Advisor (STA) performs a safety

assessment for any event or condition that places the plant in an abnormal situation or for any plant parameter that affects or reflects an abnormal indication of a safety related system. This safety assessment is intended to help ensure abnormal plant conditions are properly assessed against reportability criteria. The STA assists the Shift Supervisor in making prompt verbal reportability determinations. The Shift Supervisor determines whether the condition requires verbal reporting to a governmental agency using guidance in Standing Order R-11, *Notification of Significant Events*. Standing Order R-11 is used to perform a 10 CFR 50.72, 10 CFR 20.2201, 10 CFR 20.2202 or 10 CFR 26.73 notification to the NRC. For events reportable under 10 CFR 73.71, appropriate Security management personnel review the event in accordance with Security Administrative Procedure SAP-35, *Reporting of Safeguards Events*, and Standing Order R-12, *Reporting of Physical Security Events*, to determine reportability.

If further evaluation is needed to determine reportability, appropriate management personnel are asked for assistance. In general, procedure PED-QP-19, *Evaluation of Potentially Reportable Conditions*, is used by Engineering in determining if a reportable condition exists. This procedure guides the user through an evaluation of the condition against reporting criteria of 10 CFR 50.72, 10 CFR 50.73, 10 CFR 21, and 10 CFR 50.9(b). Completion of the PED-QP-19 documentation results in a recommendation from Engineering that a condition meets or does not meet any of the reporting criteria. This recommendation is then reviewed by the PRC, which subsequently makes a reportability recommendation to the Manager - FCS. The Manager - FCS makes the final determination of reportability and ensures that required reports are made.

Written reports are prepared in accordance with NOD-QP-35, *Licensee Event and Special Reports*, and Standing Order R-3, *Reportable Occurrences*. These procedures ensure that OPPD complies with the regulatory guidance on written reports promulgated in 10 CFR 20, 10 CFR 50.73, 10 CFR 73.71, 10 CFR 50.36 and NUREG-1022. The reporting of issues subject to 10 CFR 21 is addressed in NOD-QP-12, *Reporting of Defects and Noncompliance to the Nuclear Regulatory Commission* (10 CFR 21).

D.5 Condition Reports - Management Review

After the initial operability and reportability reviews are complete, the CR is prepared for management review by the Corrective Action Group (CAG). The CAG performs a preliminary investigation on the condition and determines whether the condition is repetitive, indicative of an adverse trend, or a generic concern. The CAG makes a preliminary determination of the significance level and assigns trend codes. This preparation enables management to complete a more informed, accurate, and timely review.

Management review of new reports in the CR process is performed during normal work day meetings by the Condition Review Group (CRG). The CRG is chaired by the Manager - FCS (or alternate) and includes department managers. Either the Vice President or one of the Division Managers (or designated alternate) normally attends the CRG meeting. The various department managers provide a multi disciplinary review team. The key purpose

of the meeting is to focus on safety, significance, operability, generic implications, reportability, and assignment of an owner.

The CRG assigns to each Condition Report a significance level dependent on: the impact to quality, reliability or safety; the recurrence of the condition; and actions already taken to correct the condition.

If subsequent evaluation of the condition reveals new information that may affect reportability, operability, or safety significance, the Manager - FCS or Shift Supervisor is notified and the CR is again presented to management at the CRG. The CRG may reconsider the CR significance level recommended by the CAG or as previously set by the CRG.

In accordance with Standing Order R-2, assigned CR owners determine the cause of the condition, develop the corrective actions, and set the schedule for completion. Only selected CRs are subject to a Root Cause Analysis. Any condition determined by the CRG to be significant must meet the requirements of 10 CFR 50 Appendix B, Criterion XVI (understanding the cause of the event and establishing corrective actions to prevent recurrence) through assignment of a Root Cause Analysis.

D.6 Root Cause Analysis

Prior to 1989, a formal and procedurally driven process for the determination of causes of events did not exist at FCS. The completion of Safety Enhancement Program Items 2, 10, and 15, resulted in Quality Procedures NOD-QP-19, *Root Cause Analysis*, and NOD-QP-20, *Human Performance Enhancement System*.

The goal of the Root Cause Analysis (RCA) program implemented through NOP-QP-19 is to provide a methodology for evaluation and analysis of selected incidents or conditions to determine root cause(s) and generic implications. Currently, OPPD requires completion of RCAs for events or conditions considered significant.

Although the implementation of the RCA program under NOD-QP-19 includes the application of Human Performance Enhancement System (HPES) process techniques, the HPES program can also be applied as a stand alone process to non-significant conditions or events through use of NOD-QP-20.

The scope of RCA program includes guidance on analytical methods and techniques, corrective action development, report preparation and approval, trending, and qualification requirements for individuals performing RCA Evaluations. Analytical methods include the use of the various MORT analysis techniques, as well as Kepner-Tregoe problem solving techniques and HPES methodologies. The RCA process relies on the use of qualified individuals throughout the nuclear organization to perform cause and generic implication determinations.

A Peer Review Team assesses each RCA report for adequacy in determining root, contributing causes and generic implications. Based on the identified cause(s), the Condition Report Owner then determines corrective actions and presents the RCA to the PRC for final approval. The PRC provides a final assessment of the adequacy of the causal determinations, generic implications, and corrective actions for the condition under investigation. The causal determinations and corrective actions are then entered into the CR as a means of providing trending data and corrective action tracking.

The CRG may, at its discretion, require a root cause analysis for nonsignificant conditions when a formal method of cause determination is desirable. Otherwise, the (apparent) cause of the nonsignificant events and appropriate corrective actions are determined by the CR owner.

D.7 Corrective Actions

The owner of a CR assigns action items to appropriate personnel as a tracking vehicle for the corrective actions. Timeliness for implementation of corrective actions is maintained by raising the level of management approval for due date extensions. Manager - FCS or Division Manager approval may be required for an extension, depending on the length of time for implementation and the priority level of the action item.

When the owner has confirmed completion of the corrective actions, the CAG will perform an administrative review for CR closure, ensuring that all required corrective action documentation is in the CR record file.

D.8 Trend and Search Capabilities

An added benefit of the CR system is the enhanced search and trending capabilities. The low threshold of reporting allows for analysis and trending that can identify precursors before serious problems occur. This provides valid information that allows for proper prioritization of resources. When the process measurement reveals ineffective corrective actions through repetitive problems, the recurrent issue can be given a higher priority or resources.

Available reports address topics such as whether the condition was self-identified, causal factors, work groups involved with the condition, and event symptoms involved. The process measurement not only identifies effectiveness of corrective actions, but also can provide data for quarterly trend reports and self-assessment. Personnel now have the ability to electronically view trend information relevant to their functional area without the need for extensive manual tabulation.

The improved search capabilities enable the CAG to provide the CRG with an accurate presentation of condition trends and history, identifying recurring events that need further corrective actions directed at the cause. Trending reports are periodically presented to both the PRC and the Safety Audit and Review Committee.

The ability to provide timely data on reported conditions has proven valuable, allowing early compensation and correction of activities in the field. For example, significant and repetitive CRs during the 1996 refueling outage were compared with Incident Reports from the 1995 refueling outage. A prominent shift was seen from industrial safety and maintenance events to equipment operational events. Significant events in the 1996 refueling outage occurred during active manipulation of installed equipment instead of during times a system was out of service for testing or maintenance. A possible explanation for this trend was scheduling problems, including a lack of system windows. This shift in events prompted immediate concern by the Manager - FCS. At his direction, a site-wide briefing was conducted to communicate this trend to the workforce. Although these problems could not be immediately corrected, the site wide briefing provided a heightened awareness and a sensitivity to safety in all aspects of the outage.

From September 1995, when the Condition Reporting System was initiated, through December 1995, approximately 2100 Condition Reports (CR) were initiated. Out of this total, 126 have been classified as configuration discrepancies. These have been further classified as plant document, design document, and physical discrepancies. A plant document discrepancy is a plant document, typically a procedure, not consistent with the design or licensing basis. A design document discrepancy is a design document not consistent with the design or licensing basis. A physical discrepancy is a plant system, structure or component not conforming to the design basis. The following table shows a breakdown of the CRs by type and cause.

	Type of Discrepancy			
Discrepancy Cause (definitions on next page)	Plant Documents	Physical	Design Documents	TOTAL
Cause Unknown	4	9	2	15
Error Updating Plant Documents for Configuration Change	15			15
Unauthorized Configuration Change		11		11
Discrepancy Due To Original Equipment		17		17
Old Modification Caused Discrepancy	3	13	5	21
Recent MR or ECN caused Discrepancy		3		3
Maintenance Activity Caused Discrepancy		12		12
Error in Original Documentation			11	11
Error in Updating Design Documentation for Configuration Change			17	17
Original Building Documents			4	4
TOTAL	22	65	39	126

The definitions of the causes are:

1. Cause Unknown - The investigation of the discrepancy was unable to identify a cause.
2. Error in Updating Plant Documents for Configuration Change - A plant document was not updated for a configuration change or an error was made when the document was revised. The configuration change was made using the OPPD's revised design engineering and configuration control processes implemented following the SSOMI inspection.
3. Unauthorized Configuration Change - A configuration change was made without the proper documentation or an unauthorized change in the construction work order was made.
4. Discrepancy Due To Original Equipment - A deviation between the original design basis and the as-built plant was identified and no configuration changes could be identified which would have modified the as-built plant.
5. Old Modification Caused Discrepancy - A modification was incorrectly installed using the design engineering and configuration control processes in place prior to the SSOMI inspection.
6. Recent MR or ECN Caused Discrepancy - A modification or configuration change was incorrectly installed using the design engineering and configuration control processes implemented following the SSOMI inspection.
7. Maintenance Activity Caused Discrepancy - A configuration change was made using the Maintenance Work Order process.
8. Error in Original Documentation - A deviation between the original design basis documents and the as-built plant was identified and no document changes could be identified which would have modified the design basis documents
9. Error in Updating Design Documentation for Configuration Change - A design document was not updated for a configuration change or an error was made when the document was revised. The configuration change was made using the OPPD's revised design engineering and configuration control processes implemented following the SSOMI inspection.
10. Original Building Documents - Errors or omissions are identified in documentation for non-safety structures (e.g., warehouse) constructed using commercial grade standards.

D.9 Audits and Inspections of the CAP

Quality Assurance Audit #45 of the CAP includes verification that conditions adverse to quality are effectively identified, documented and corrected in a timely manner. The 1994 and 1995 audits determined that the CAP was effectively implemented but that there was an excessive number of overdue responses. The auditors also noted an attitude among personnel which encouraged rapid rather than complete corrective actions. Operational non-conformance packages reviewed provided adequate technical resolution to the identified problems.

The 1996 audit determined that the CR system was exhibiting consistency and uniformity. Although not all administrative problems were completely resolved, the system was verified to be effective in completing corrective actions. Strengths noted were effectiveness of the CRG in defining a problem, assigning ownership and determining significance level; excellent CAG presentations; and management's lower threshold of problem identification.

The 1996 FCS Joint Utility Management Audit report noted an ongoing problem with timeliness of responses and deficiency correction. The timeliness of corrective action responses was found to need improvement. To improve responsiveness and compliance with due dates, a daily status report is available to the Vice-President and to managers responsible for condition reports. Improved compliance with commitment dates for response is being enforced by all levels of management.

The 1994 NRC Systematic Assessment of Licensee Performance (SALP) report for FCS (Inspection Report 94-99) noted a weakness in knowledge about the CAP on the part of station personnel. Personnel were confused as to which system was appropriate for use and which problems should be reported. These were key reasons for the upgrade of the CAP to the CR system. In 1996, the NRC SALP Report 96-99 noted that personnel knowledge of the CAP had improved to a good level. The report also notes that plant personnel were using the CAP to document problems including minor configuration discrepancies. Training programs had been updated to include methods for identifying problems.

Engineering was noted as being effective in identifying, resolving and preventing problems in the 1995 NRC Integrated Performance Assessment Process (IPAP) report for FCS (Inspection Report 95-18). The IPAP report also noted FCS was effective in identifying and resolving performance problems and cited strong problem ownership at FCS as a key to resolution of issues and correction of deficiencies.

D.10 Conclusion

The current Corrective Action Program provides an effective process for problem identification, operability and reportability determination, implementation of corrective actions, and trending. Based on the increased number of CRs and the independent assessments the CAP is meeting management expectations of a low threshold for reporting of problems. However, cause determination and corrective action determination and completion for some CRs are not being completed in accordance with the due dates. OPPD has initiated corrective action to resolve this issue, including reinforcement of management expectations and establishment of additional tracking to ensure completion of actions in accordance with the due dates.

Omaha Public Power District Response to NRC Request E

(E) The overall effectiveness of your current processes and programs in concluding that the configuration of your plant is consistent with the design bases.

Executive Summary

The Omaha Public Power District (OPPD) processes, procedures and programs provide reasonable assurance that the Fort Calhoun Station (FCS) physical configuration is consistent with the design bases. This assurance is based on:

1. The reconstitution and verification of the FCS design bases completed in 1990 (Section E.1)

2. The confirmation that the plant physical configuration conformed to the design bases following completion of the reconstitution of the design bases (Section E.2)

3. The verification of the Updated Safety Analysis Report (USAR) to be consistent with the design bases (Section E.3)

4. The completion of the Procedures Upgrade Project and the associated Design Bases/Safety Related Operating Procedure Compliance Project (Section E.4)

5. The use of the Operating Experience Review Process and the Commitment Action Tracking System to ensure that OPPD complies with new NRC requirements (Section E.5)

6. The timely identification and resolution of configuration discrepancies using the Corrective Action Program (Section E.13)

7. The participation of the system engineers in the design change and configuration control process, ensuring effective implementation of configuration control procedures and compliance with the design basis (Section A.2)

8. The ready availability of the USAR, Technical Specifications, Design Basis Documents and other licensing documents (Section E.6)

9. The upgrade of the processes, programs and procedures for changing and controlling the configuration of the station as part of the Safety Enhancement Program in conjunction with the Design Basis Reconstitution (Section E.9)

10. The completion of vertical slice SSFI and SSOMI type audits since the deployment of the upgraded procedures and Design Basis Reconstitution which have confirmed these processes were and are being effectively implemented (Sections E.7 and E.10)

11. The results of inspections by the NRC of the reconstitution of the design basis and the processes, programs, and procedures for changing and controlling the configuration of the station (Sections E.8 and E.11)

12. The additional verification of the USAR and design bases currently being conducted by OPPD which have identified areas for improvement, but have confirmed the overall effectiveness of processes, programs and procedures for maintaining conformance with reconstituted design bases (Section E.12)

E.1 Design Basis Reconstitution

The FCS design bases were reconstituted and conformance of the as-built plant with the design bases was verified following the NRC Safety System Outage Modification Inspections in 1985. The Design Basis Reconstitution Project completed development of the Design Basis Documents in April 1990 as described in the response to Request C. This reconstitution included an evaluation of modifications installed on safety related systems since the initial operation of the station. The Open Item Program, described in the response to Request C, was used to identify, evaluate, classify the safety significance, and resolve design bases discrepancies. These discrepancies included missing design documents, inadequate design documents, and documents containing conflicting information. These open items ranged from minor discrepancies, which had no impact on the safety of the plant, to issues which potentially required notification of the NRC.

To ensure the design bases were maintained during the design basis reconstitution, OPPD adopted the following interim policy for assessment of configuration changes to FCS:

> *The Fort Calhoun USAR in conjunction with any pending changes to Fort Calhoun design drawings and Safety Evaluation Reports used for amendments to the technical specification shall be used for providing design bases for activities such as modifications, procedure changes, and safety evaluations.*

In conjunction with the early phases of the design bases reconstitution, the OPPD engineering, design, and configuration control processes were upgraded to ensure that subsequent changes were effectively controlled and the changes were verified to conform with the design bases.

OPPD has completed internal Safety System Functional Inspections (SSFIs) on 8 systems. The intent has been to validate the effectiveness of the Design Basis Reconstitution Program by assessing the ability of the systems to meet their design bases. These SSFIs have confirmed the operational readiness of the systems and the effectiveness of corrective actions implemented as part of the SEP.

The Design Basis Reconstitution Project and closure of DBD Open Items successfully recovered and reconstituted the FCS design bases. This Project consolidated the design basis into the Design Basis Documents (DBDs). Independent confirmation that the design basis was successfully reconstituted is discussed in Section E.7.

E.2 Design Bases Verification

Verification that the FCS physical configuration conformed to the reconstituted design bases was accomplished by physical verification, functional verification and a safety evaluation check.

1. The design requirements were compared against the current plant configuration as shown on controlled documents (i.e., drawings, specifications, etc.).

2. Walkdowns were conducted to verify that the controlled documents conformed with the physical plant.

3. Modifications implemented by OPPD after issuance of the FCS operating license were reviewed to include any new design requirements, and the associated 10 CFR 50.59 safety evaluations were reviewed to verify their adequacy. Concerns identified during this review were tracked and resolved by the DBD Open Item program.

4. The functional requirements identified in the DBDs were compared against existing OPPD procedures to insure surveillance and in-service testing was being adequately performed to verify system and component operability over time.

Additional actions included physical walkdowns of plant components to verify the maintenance data base and an upgrade of the vendor manuals to incorporate updated vendor information. Procedures controlling both the maintenance data base and the vendor manuals were also upgraded to ensure both the data base and vendor manuals received timely and appropriate updating.

Details of the Design Bases Verification are included in the response to Request C. These verification actions provide adequate confidence that the physical plant conforms with the completed Design Basis Documents. The independent verification of this project, discussed in Section E.6, provided assurance that this project achieved its objective.

E.3 USAR Verification

The USAR Verification Project, described in the response to Request C, verified that the information contained in the USAR conformed to the information gathered during the Design Basis Reconstitution Project into the DBDs. The USAR Verification Project also verified that a proper review and inclusion of information required by 10 CFR 50.71(e) had been completed. The Project also revised the USAR update and review procedures as necessary to maintain the USAR information current and consistent with the design bases.

E.4 Procedures Upgrade Projects

FCS procedures were rewritten/upgraded in the late 1980s through the early 1990s by completion of the Procedures Upgrade Project and the associated Design Bases/Safety Related Operating Procedure Compliance Project. Details of these projects are discussed in the response to Request B. Based on the successful completion of these projects, OPPD has reasonable assurance that the FCS design bases are reflected in the current operating, maintenance and testing procedures.

E.5 Operating Experience Review (OER) and Commitment Action Tracking System

The FCS OER Program and Commitment Action Tracking System are used to evaluate and address NRC requirements and generic information distributed in the form of Generic Letters, Bulletins, Information Notices, Administrative Letters, changes to regulations, and applicable 10 CFR Part 21 Reports. The OER program was established to provide evaluations of industry operating experience, approve corrective actions, and assure implementation of corrective actions. Each item is screened to determine potential applicability, significance and priority. The recommendations are tracked to completion in the Commitment Action Tracking System. Deficiencies considered to be adverse to quality are addressed through the Condition Report corrective action process.

An example of this process is the OPPD response to the requirements of Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities. As a result of these requirements, OPPD has completed various walkdowns, upgrades to physical structures and revisions to design basis documentation over the last several years.

The response to OER item may be classified as an Ongoing Commitment. Nuclear Operations Division Quality Procedure NOD-QP-34, *Ongoing Commitment Program*, provides administrative controls for processing and maintaining such commitments. An ongoing commitment is a docketed commitment to the NRC detailing a procedure or administrative corrective action, or a statement of intended compliance with a specific industry standard which is performed periodically or continuously and does not include an external reporting requirement. References to ongoing commitments are identified in OPPD procedures to alert the users such that the licensing bases are maintained.

The implementation of the OER Program and Commitment Action Tracking System provide adequate confidence that NRC requirements are evaluated and incorporated into the FCS design and licensing bases.

E.6 Availability of Design Bases Documentation

To assist personnel involved in design changes and safety evaluations, OPPD provides easy access to the USAR, Technical Specifications, Design Basis Documents and other licensing basis documents.

1. Controlled hard copies of the USAR, Technical Specifications, and Design Basis Documents (DBD) are maintained in the plant and the administration building.

2. Full text search capability is available on the Fort Calhoun LAN for electronic copies of the USAR, Technical Specifications, NRC Safety Evaluations of Technical Specification Amendments, OER documents, Design Basis Documents and procedures. Although these are not "controlled" copies, they are restricted such that they may only be updated by authorized personnel.

These actions provide reasonable assurance that personnel have ready access to the DBDs, USAR, Technical Specifications and other design/licensing bases documents.

E.7 Assessments of Design Bases

Since 1987, OPPD has performed 9 (including one follow-up) audits based on NRC guidance for Safety System Functional Inspections (SSFIs). During these audits, vertical slice evaluations were performed on 8 systems: Auxiliary Feedwater, Instrument Air, 120 VAC Vital Distribution, Component Cooling Water, 125 VDC, Raw Water, Stator Cooling, and Diesel Generator. The purpose of these audits was to verify the operational readiness of the systems, including system modifications, testing, maintenance and operation, to perform required safety functions in conformance with the current licensing bases. These audits also served to validate the effectiveness of the Design Basis Reconstitution Program.

The follow-up SSFI audit was performed to ensure that corrective actions were performed for deficiencies identified during the Instrument Air, Component Cooling Water, and Auxiliary Feedwater SSFI audits. This follow-up audit also included verifications that deficiencies identified by OPPD prior to the NRC Electrical Distribution System Functional Inspection (EDSFI) were corrected or on track for correction.

The subsequent SSFIs each generated approximately three concerns that dealt with items such as inconsistencies between USAR values and calculations, inadequate post-maintenance testing of equipment, and administrative deficiencies with Design Basis Documents, plant procedures, and drawings.

These audits show a continuing trend of improved conformance of the plant to its design bases. Review of corrective action documents generated as a result of these inspections noted that corrective actions such as updates to procedures, System Training Manuals and drawings, changes to operation and testing procedures, and corrections to the Computerized History and Maintenance Program System (CHAMPS) equipment database have taken place to correct the identified deficiencies.

During the months of October and November 1994, OPPD personnel conducted the Service Water System Operations Performance Inspection (SWSOPI). This assessment determined that the Raw Water, Component Cooling Water and related supporting systems are in good condition. The assessment also concluded that the systems are operated within the design criteria for FCS.

The assessment activities described above provide OPPD with reasonable assurance that the configuration and performance of the FCS systems, structures and components are consistent with the design bases.

E.8 NRC Inspections of Design Bases

In addition to assessments conducted by OPPD which evaluated effectiveness of the FCS Design Basis Reconstitution Project and configuration control process, the NRC has conducted several major inspections of OPPD's programs. In April 1990, the NRC conducted an Electrical Distribution System Functional Inspection (EDSFI), as documented by Inspection Report 91-01, dated May 20, 1991. The inspection team reviewed selected modifications, design calculations, and design basis documents. Additionally, the inspection team conducted interviews and reviewed training records, corrective action records, maintenance records and surveillance test records. The review focused on the off-site power supply grid and associated on-site, safety-related busses and loads. During the EDSFI, the team noted a number of strengths and relatively few weaknesses. The team noted that the design basis documents were generally comprehensive, user friendly, and provided a good information source to the engineering staff. Design information was readily retrievable and accurate.

An NRC/NRR Special Audit of Control Processes for Commitments and Current Licensing Basis was documented in a report dated October 19, 1993. The staff conducted audits at a cross section of reactor plants to assess the processes used by licensees for controlling commitments that affect the plants' current licensing basis, including maintaining and updating the final safety analysis report. At FCS, the audit team found that commitments which affected the USAR were captured by the USAR update process. The team's review of several plant modifications to identify changes to plant systems and verify incorporation of the changes in the USAR showed that all affected text descriptions and system drawings in the USAR were properly revised to reflect the associated modifications.

An inspection of OPPD's Engineering and Technical Support (E&TS) capability was completed August 25, 1995, as documented by Inspection Report 95-11, dated October 19, 1995. This inspection was conducted to evaluate the ability of the Production Engineering Division to provide effective engineering and technical support to the plant. In addition to inspecting the activities of the engineering group, the inspectors also reviewed the processing of 10 CFR 50.59 safety evaluations. The team concluded that Design Engineering was accomplishing its goals and managing workloads. The team also noted that OPPD's response to an industry event involving pipe rupture because of a less than adequate erosion/corrosion program was exemplary. The team found the effectiveness of the Nuclear Safety Review Group to also be a strength. The inspectors identified that a plant modification was implemented using a process that specifically does not allow such changes to the plant design basis. The team considered this a failure to perform a 10 CFR 50.59 safety evaluation. Since this finding, OPPD has implemented extensive enhancements to the 10 CFR 50.59 process.

The inspection activities described above provide OPPD with reasonable assurance that the configuration and performance of the FCS systems, structures and components are consistent with the design bases.

E.9 Engineering Design and Configuration Control

Engineering design control is conducted in accordance with 10 CFR 50 Appendix B, Criterion III; ANSI N 45.2.11; 10 CFR 50.59; and 10 CFR 50.71 as described in detail in the responses to Requests A and C. Key provisions are listed below.

1. Design work is reviewed for conformance with the design bases or changes are made to the bases.

2. The Design Change Package (DCP) and supporting material (calculations, analyses, specifications, drawings, etc.) are developed using approved procedures and are maintained as controlled documents.

3. Design verification is conducted in accordance with established procedures.

4. System interaction analysis is used to assess the effects of a design change system requirements.

5. Plant and Engineering management approval processes are defined.

6. The impact of a design change on plant procedures and training is assessed.

7. Changes to the USAR, Technical Specifications, and procedures and modifications are reviewed.

8. Applicable procedure and design changes are screened to determine if changes to the USAR or Technical Specifications are required.

9. The review and approval process for 10 CFR 50.59 evaluations is defined.

10. A process is in place for determining the information to incorporate into the USAR.

Thus, the FCS configuration control processes provide reasonable assurance that the plant's approved configuration is maintained.

E.10 Assessments of Engineering Design and Configuration Control Processes

Internal Quality Assurance audits have evaluated the engineering design and configuration control processes during both normal and refueling operations. These were implemented as Safety System Functional Inspections and a Safety System Outage Modification Inspection.

Additionally, Quality Assurance personnel perform periodic audits of both the Engineering Configuration Management program and the Station Engineering program. To support the established audit program, Quality Assurance has established a surveillance program that periodically evaluates the engineering activities with potential to affect the Licensing Bases. These surveillances encompass activities associated with Station Engineering, Modifications Performed During Outages, Reload Analyses, Design Basis/Drawing Control, the 10 CFR 50.59 safety evaluation program, and fire protection. Formal review of modification close-out is performed during the Quality Assurance audit of engineering configuration management. This review ensures that design bases documents that are required to be updated as a result of station modification have been updated and that modifications that are performed are done such that the licensing bases are maintained.

Review of audit reports related to Engineering Configuration Management noted that the Design Change Administration Audit, conducted in 1987 and 1988, reviewed approximately 70 modification packages. Based upon these reviews, deficiencies were identified that included: lack of design basis information, lack of engineering judgment/assumption documentation, inadequate work instructions, incomplete construction work packages, insufficient 10 CFR 50.59 safety evaluations, and lack of administrative control for the training program for engineers. Subsequent audits noted that corrective actions had been effectively implemented and audit personnel who had participated in the previous audit noted improved modification package quality. The Design Change Administration Audit, performed in 1991, determined that previous audit findings had been corrected. Two corrective action reports were generated during this review. These reports were administrative in nature and dealt with completion of documentation reviews.

During the years from 1990 to 1993, QA personnel further verified compliance with the Appendix B Criterion III by performing the Design Engineering Audit. This audit was performed on a yearly basis, covering the four disciplines of Engineering. These audits verified that, even though some isolated conditions adverse to quality were identified, the engineering design process was effectively implemented.

The 1995 configuration processes audit identified weaknesses in the areas of documentation update, engineering services procurement, procedure quality and QA program compliance. These conditions adverse to quality were considered isolated and no programmatic problems were identified.

A review of the QA surveillance reports notes that very few conditions adverse to quality were identified during the modification reviews. Over the past four refueling outages (1992 through 1996), approximately 65 modifications have been reviewed. Nine (9) corrective action documents were issued to document concerns. None of the identified weaknesses were considered to indicative of a systematic or programmatic problem.

Review of these audits revealed that programs have been established and improved to ensure modifications performed at FCS that maintain the approved design and licensing bases. Concerns identified during the early audits have been corrected by enhancing Engineering Instructions and providing necessary training to Engineering personnel.

The OPPD Nuclear Safety Review Group provides a monthly review of safety evaluations conducted in accordance with 10 CFR 50.59. These reviews have documented a steady improvement in the overall quality of these evaluations. The 1996 reviews show the number of "attention to detail" type errors have been reduced during the year.

The OPPD internal assessments provide reasonable assurance that the design change and configuration control processes are effectively implemented and maintain the physical configuration of the plant in conformance with the design bases.

E.11 NRC Inspection of Engineering Design and Configuration Control Processes

Several NRC inspections have assessed the FCS engineering design and configuration control processes. A 1992 inspection (92-04) included the following statement:

> *The team concluded that the licensee had developed and implemented a program to control design changes and modifications in accordance with the TS, the USAR and regulatory requirements. The team considered the performance of safety evaluations to be exceptional.*

The same team also concluded that the temporary modification program effectively controlled temporary modification in accordance with the Technical Specifications, USAR and regulatory requirements.

A 1993 inspection report (93-08) affirmed that FCS had a good program for performing safety evaluations in accordance with 10 CFR 50.59. The same inspection noted examples of incomplete evaluations, evaluations that lacked attention to detail, and one screening that was not adequate. A 1995 inspection report (95-11) also noted that the lack of attention to detail on the part of reviewers of 10 CFR 50.59 safety evaluations. OPPD initiated actions to improve the overall quality of safety evaluations and attention to detail. The quality of the safety evaluations is evaluated and tracked by the Nuclear Safety Review Group as discussed in the previous section. The trend shows an improvement in the quality of evaluations since the corrective actions were implemented in early 1996.

A 1994 maintenance reliability initiative inspection report (94-09) noted that the FCS procedures for modification control were comprehensive and provided an in-depth description of responsibilities and detailed instructions. The inspection team evaluated two modifications to determine effectiveness of OPPD's implementation of these procedures. The team concluded the procedures were effectively implemented.

A 1995 inspection report (95-11) identified several minor inaccuracies between the physical configuration, the DBDs and the USAR. The team determined OPPD had identified these problems and had good programs to correct the problems. The team concluded that the programs to correct the inaccuracies in the DBDs and USAR appeared to be effective and good progress was being made to reduce the number of these inaccuracies.

E.12 Ongoing Verification Activities

On a quarterly basis, System Engineers develop report cards providing commentaries on the health of their respective systems. These reports also identify any open modifications, Engineering Assistance Requests, and other configuration control issues.

In order to provide additional assurance of conformance with the design bases, OPPD is currently completing assessments of selected FCS systems using the Nuclear Energy Institute (NEI) *Guidelines for Assessing Programs for Maintaining the Licensing Basis* (NEI 96-05). Deficiencies found are entered into the Condition Report system for resolution. The assessment of the Chemical and Volume Control System (CVCS) is complete, and assessments of the Safety Injection and Instrument Air systems are ongoing. OPPD will complete assessments of all remaining FCS safety-related and safety-significant systems no later than February 1, 1999.

A comparison was made between information in USAR Section 9.2 (CVCS) and information in DBD #108 (CVCS). Approximately 90% of the USAR text was reviewed and verified consistent with text in the DBD. Overall, the results of the comparison were positive. To a large degree, the USAR and DBD describe information related to the CVCS in a consistent manner, and the information corresponds to the plant physical configuration. Three minor discrepancies were found during this comparison effort.

Conclusions of the CVCS assessment included:

- No licensing basis discrepancies that could result in a significant public health or safety concern were identified during the course of this self assessment.

- One instance was found where the plant had not been operated in accordance with the Technical Specifications. Licensee Event Report 96-006 is being prepared to report that all charging pumps were rendered inoperable for approximately a 10 hour period in violation of the Technical Specification requirements (CR 199601087).

- Several items were identified that would improve the maintenance of the licensing basis:

  1. Process improvements are needed for initiating and maintaining Technical Specification Interpretations.
  2. Programmatic guidance is needed to assess the need to incorporate current operating practices, i.e. operations memos, operator workarounds, old tag-outs, temporary modifications, old nonconformances, into the licensing basis documents.
  3. Procedural guidance is needed for evaluating QA Program changes for reduction in commitment to the NRC prior to implementation.
  4. Process improvements are needed for more timely update of the USAR following configuration changes to the station. It was noted that a more efficient modification acceptance process is currently under development.

The identified reportable condition and the areas for improvement were entered into the Corrective Action Program discussed in the response to Request D.

E.13 Condition Report System Trend

The Condition Report system discussed in the response to Request D provides OPPD with the ability to effectively track plant configuration discrepancies. The Corrective Action Program is being effectively used to identify configuration control deficiencies and correct these deficiencies. The large majority of the configuration discrepancies are minor in nature, as defined in Section D.8.

E.14 Conclusion

The OPPD processes, procedures and programs described in the responses to Requests A through E provide adequate confidence that the FCS physical configuration is maintained consistent with the design bases, and that design bases requirements are translated into operating, maintenance, and testing procedures. The results of assessments by OPPD and NRC confirm this conclusion.

LIC-97-011
Attachment 3

Acronym Definitions and
Referenced Procedures

Acronym Definitions

Acronym	Definition
ALARA	As Low As Reasonably Achievable
AOP	Abnormal Operating Procedure
CAG	Corrective Action Group
CAP	Corrective Action Program
CAR	Corrective Action Report
CAS	Central Alarm Station
CE	Combustion Engineering
CHAMPS	Computerized History and Maintenance Program System
CR	Condition Report
CRG	Condition Review Group
CVCS	Chemical and Volume Control System
CWO	Construction Work Order
DBD	Design Basis Document
DBR	Design Basis Reconstitution
DC-ECN	Document Change Engineering Change Notice
DCP	Design Change Package
E&TS	Engineering and Technical Support
EAR	Engineering Assistance Request
EDSFI	Electrical Distribution System Functional Inspection
EOP	Emergency Operating Procedure
ESP	Engineering Support Personnel
FC-ECN	Facility Change Engineering Change Notice
FCS	Fort Calhoun Station
FLC	Facility License Change
GEI	General Engineering Instruction
GL	Generic Letter
HPES	Human Performance Enhancement System
IEB	Inspection and Enforcement Bulletins
IET	Incident Evaluation Team
INA	Independent Nuclear Assessment
IPAP	Integrated Performance Assessment Process
IR	Incident Report
ISD	Instructional Systems Design
JCO	Justification for Continued Operation
LER	Licensee Event Report
MR	Modification Request
MWO	Maintenance Work Order
MWR	Maintenance Work Request
NEI	Nuclear Energy Institute
NOD-QP	Nuclear Operations Division Quality Procedure

NSRG	Nuclear Safety Review Group
NSSS	Nuclear Steam Supply System
OER	Operating Experience Review
ONC	Operational Non-Conformance
P&ID	Piping & Instrumentation Diagram
PED-GEI	Production Engineering Division General Engineering Instruction
PED-QP	Production Engineering Division Quality Procedure
PLDBD	Plant Level Design Basis Document
PRC	Plant Review Committee
PUP	Procedures Upgrade Project
QA	Quality Assurance
QAM	Quality Assurance Manual
QC	Quality Control
QP	Quality Procedures
QR	Qualified Reviewer
RCA	Root Cause Analysis
SAC	System Acceptance Committee
SALP	Systematic Assessment of Licensee Performance
SAO	Safety Analysis for Operability
SARC	Safety Audit and Review Committee
SDBD	System Level Design Basis Document
SEP	Safety Enhancement Program
SER	Safety Evaluation Report
SMART	Station Modification Acceptance and Review Team
SO	Standing Order
SR	Safety Related
SRI-ECN	Substitute Replacement Item Engineering Change Notice
SSC	Structure, System or Component
SSFI	Safety System Functional Inspection
SSOMI	Safety System Outage Modification Inspection
SWSOPI	Service Water System Operational Performance Inspection
TDB	Technical Data Book
TM	Temporary Modification
USAR	Updated Safety Analysis Report
USQ	Unreviewed Safety Question

Referenced Procedures

Procedure No.	Procedure Title
NOD-QP-3	10 CFR 50.59 Safety Evaluations
NOD-QP-7	Facility License Changes (FLCs)
NOD-QP-12	Reporting of Defects and Noncompliance to the Nuclear Regulatory Commission
NOD-QP-16	Updated Safety Analysis Report (USAR)
NOD-QP-19	Root Cause Analysis Program
NOD-QP-20	Human Performance Enhancement System Program
NOD-QP-22	Preparation and Approval of a Safety Analysis for Operation (SAO)
NOD-QP-31	Operability and Reportability Determinations
NOD-QP-34	Ongoing Commitment Program
NOD-QP-35	Licensee Event and Special Reports
PED-GEI-3	Preparation of Design Change Packages
PED-GEI-29	Facility Change Evaluation
PED-GEI-35	Preparation of EARs for Minor Configuration Changes/Replacements
PED-GEI-60	Substitute Replacement Item Evaluations
PED-QP-2	Configuration Change Control
PED-QP-3	Calculation Preparation, Review and Approval
PED-QP-5	Engineering Analysis Preparation, Review and Approval
PED-QP-19	Evaluation of Potentially Reportable Conditions
PED-QP-29	Evaluating, Reconstituting, and Closing Design Basis Document Open Items
QAM-14	Biennial Procedure Review Oversight Program
SAP-35	Reporting of Safeguards Events
SO-G-21	Modification Control
SO-G-30	Procedure Changes and Generation
SO-G-73	Fort Calhoun Station Writer's Guide
SO-M-101	Maintenance Work Control
SO-O-25	Temporary Modification Control
SO-R-2	Condition Reporting and Corrective Action
SO-R-3	Reportable Occurrences
SO-R-11	Notification of Significant Events
SO-R-12	Reporting of Physical Security Events