CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9702190315 DOC.DATE: 97/02/10 NOTARIZED: YES DOCKET #

FACIL:		DOCKET #
50-269	Oconee Nuclear Station, Unit 1, Duke Power Co.	05000269
50-270	Oconee Nuclear Station, Unit 2, Duke Power Co.	05000270
50-287	Oconee Nuclear Station, Unit 3, Duke Power Co.	05000287
50-369	William B. McGuire Nuclear Station, Unit 1, Duke Powe	05000369
50-370	William B. McGuire Nuclear Station, Unit 2, Duke Powe	05000370
50-413	Catawba Nuclear Station, Unit 1, Duke Power Co.	05000413
50-414	Catawba Nuclear Station, Unit 2, Duke Power Co.	05000414

AUTH.NAME	AUTHOR AFFILIATION
TUCKMAN,M.S.	Duke Power Co.
RECIP.NAME	RECIPIENT AFFILIATION
	Document Control Branch (Document Control Desk)

See Reports

SUBJECT: Provides response to RAI re adequacy & availability of design basis info.

DISTRIBUTION CODE: A074D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 78
TITLE: Responses to 50.54(f) Req. for Design Basis Info

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD2-2 PD	1	1	LABARGE,D	1	1
	NERSES,V	1	1	TAM,P.	1	1
INTERNAL:	~~FILE CENTER 01~~	1	1	NRR/DRPM/PGEB	1	1
	~~PDR-2/THOMAS,K~~	3	3			
EXTERNAL:	NRC PDR	1	1			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 10 ENCL 10

MA

Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

M. S. TUCKMAN
Senior Vice President
Nuclear Generation
(704)382-2200 Office
(704)382-4360 Fax

DUKE POWER

February 10, 1997

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Subject: Oconee Nuclear Station
Docket Nos. 50-269, 50-270, 50-287
TAC Nos. M97616, M97617, M97618
McGuire Nuclear Station
Docket Nos. 50-369, 50-370
TAC Nos. M97609, M97610
Catawba Nuclear Station
Docket Nos. 50-413, 50-414
TAC Nos. M97574, M97575

Adequacy and Availability of Design Basis Information

Reference: (1) NRC Letter dated October 9, 1996 from James M. Taylor to W. H. Grigg

Pursuant to 10CFR 50.54(f) please find attached information requested by Reference (1).

This response addresses the detailed information requested in paragraphs (a) through (e) on pages 6 and 7 of Reference (1) for Duke Power's Oconee, McGuire and Catawba nuclear stations. An Executive Summary is also included which provides a brief synopsis of the response to each of the specific information requests in Reference (1).

Appendix A to this response includes a description of the Design Basis Review program that was conducted at all three Duke Power Company nuclear stations. This program resulted in the consolidation of design basis information for each station and was a multi-million dollar program which was completed in December, 1995.

A074 1/,

It should be noted that the conclusions requested in Reference (1) cannot be stated absolutely without a detailed line by line review of each station's design and licensing basis and physical verification of each statement through plant walkdowns. This was not practical nor possible within the time constraints specified. Rather, this response should be interpreted as a good faith effort to respond to very broad questions on a limited schedule.

This response does not contain any regulatory commitments. It is primarily a look at how Duke Power currently maintains and operates its nuclear facilities with particular emphasis on assuring that the plants' design bases are maintained. Changes are made to the programs and procedures described in this response on a routine basis. These changes are part of Duke's normal process improvement to keep our programs and procedures effective and in line with current practices.

One intangible but important factor relative to understanding and maintaining the design basis for Duke's plants is that Duke Power Company has had continuity of design authority from the original licensing and design process for all three nuclear stations all the way through to the present. While Duke has been through several reorganizations, the foundation for that design authority was formed years ago and is embedded in the processes described within this document. Because of this, Duke has confidence that through the original design process and subsequent operation that the configuration management program has remained strong.

Duke Power's programs and procedures in conjunction with the oversight processes described in this letter provide reasonable assurance that the three Duke nuclear stations are operated and maintained in accordance with their design bases. When deviations or discrepancies are noted, there are processes in place to assure that timely and effective corrective action is taken to remedy any identified problem.

If there are any questions on the information contained in this letter, please contact G. A. Copp at (704) 382-5826.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

mst/gac

attachments

xc: w/att

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Luis A. Reyes
Regional Administrator, Region II

M. A. Scott
NRC Senior Resident Inspector, ONS

R. J. Freudenberger
NRC Senior Resident Inspector, CNS

S. M. Shaeffer
NRC Senior Resident Inspector, MNS

Document Control Desk
February 10, 1997

M. S. Tuckman being duly sworn states that he is Senior Vice President, Nuclear Generation Department, of Duke Power Company; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission this response to the information request filed pursuant to 10CFR 50.54(f); and that all statements and matters set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman

Subscribed and sworn to me February 10, 1997
Date

Kathy S. Moraleda
Notary Public

My commission expires: December 13, 1998

TABLE OF CONTENTS

TABLE OF CONTENTS (cont.)

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

Described below is a very brief overview of the response to each of the specific information requests contained in the NRC letter.

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

Response Summary

In general, Duke Power relies on system wide directives which describe the processes used to provide reasonable assurance that all three Duke nuclear stations are designed, operated, and maintained in accordance with their design bases. Detailed procedures for the engineering design and configuration control processes are contained in these directives. Nuclear System Directives (NSDs) cover a wide range of subjects and contain more detail than the general programs described in the Quality Assurance (QA) Topical Report. The response to information request (a) consists of a description of each of the system directives (NSDs and others) which contain procedures that either directly or indirectly affect plant configuration or plant design.

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

Response Summary

This rationale is provided by the availability of design basis information, including Design Basis Documents, through the design control and modification processes, the document control process that distributes this information, and the procedure control and review processes which are intended to assure that appropriate technical reviews including 10CFR 50.59 evaluations are performed. These processes are the subject of periodic self assessments and Duke regulatory audits. Duke's Self Initiated Technical Audits have also been conducted which have provided a vertical slice review of selected systems. The effectiveness of the operating, maintenance and testing processes as assessed by the internal oversight programs provide the requisite introspective look at how well these processes are working relative to maintaining the plants within their design bases.

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

Response Summary

The rationale for concluding that structure, system, and component (SSC) configuration and performance are consistent with the design bases relies on a similar rationale as that described in the response to (b); namely, the availability of design basis information, the design control and modification processes and the document control process that distributes design information. The baseline established by the initial startup test program, the current testing program (both post-maintenance and post-modification testing), and the periodic testing program (preventive maintenance, predictive maintenance, and Technical Specification surveillance testing) provide for ongoing assurance that plant SSCs perform in accordance with their design bases. The oversight of these processes through self assessments and audits, in particular, Self Initiated Technical Audits, provides additional assurance that the processes are effective relative to assuring that the plant SSCs conform to their design bases.

(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;

Response Summary

The response to information request (d) describes the NSDs that are included in the corrective action program. These directives are:

NSD 210 - Corrective Action Program
NSD 204 - Operating Experience Program
NSD 208 - Problem Investigation Process
NSD 212 - Cause Analysis
NSD 203 - Operability
NSD 216 - Significant Event Investigation Teams
NSD 202 - Reportability

The corrective action program provides the mechanism to assure that issues are identified, tracked, reported and resolved, as necessary.

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

Response Summary

Duke Power believes that the processes and programs in conjunction with the self assessment program described in this response are effective in providing reasonable assurance that the Duke Power nuclear stations are operated and maintained in accordance with their respective design bases.

The basis for this belief is the strong management ownership in the processes and programs and Duke Power's self assessment program, which identifies and brings to resolution process deficiencies, which enable Duke Power to correct and enhance the effectiveness of these processes. The response describes the self assessment program which includes Regulatory Audits and Self Initiated Technical Audits (SITA). When process deficiencies are identified, Duke's corrective action program provides the mechanism for necessary follow-up and corrective action.

Design Basis Review Program

Duke Power initiated a Design Basis Review program at all three nuclear stations in 1989 and completed the program in December, 1995. This program resulted in the creation of a series of Design Basis Documents (DBDs) which consolidated design basis information for a number of systems and design attributes. Appendix A contains a description of this program.

Conclusion

Duke Power's programs and procedures, in conjunction with the oversight processes described in this letter, provide reasonable assurance that all three Duke nuclear stations are operated and maintained in accordance with their design bases. When deviations or discrepancies are noted, processes are in place to provide reasonable assurance that the problems are appropriately prioritized, evaluated to determine their extent, and resolved commensurate with the level of significance.

1.0 INTRODUCTION

This response describes how Duke Power maintains and adheres to the design bases for Oconee, McGuire and Catawba Nuclear Stations, and also discusses programs for maintaining the adequacy and availability of design basis information. The Duke Power Company Quality Assurance (QA) Topical Report describes in general terms the design control and configuration control programs for nuclear safety related structures, systems, and components (SSCs). The QA Topical Report applies to nuclear safety related (QA Condition 1) SSCs. Accordingly, the focus of this response will be on design basis and configuration control processes for QA Condition 1 SSCs. However, the same processes are used in large measure on other QA Condition and non-QA SSCs. This is particularly true at Oconee Nuclear Station where the plant design takes credit for non-QA SSCs to mitigate certain design basis events.

The term "design bases," as used in this response, is defined in the same manner as in 10CFR 50.2:

> ".. that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design."

This response reflects the status of current programs and processes used by Duke Power to maintain the adequacy and availability of design basis information. These programs and processes have evolved over the years as a result of organizational changes, ongoing industry and NRC developments, and internal process improvement efforts. However, the basic intent of these programs has not changed. Duke Power has endeavored to keep abreast of the latest industry and NRC positions on design basis and configuration management programs. As an example, Duke Power has a procedure for performing 10CFR 50.59 evaluations. With the development of NSAC-125, this guidance was incorporated into Duke Power's 10CFR 50.59 directive. This directive will likely be revised as the NRC and the Nuclear Energy Institute resolve the current issues regarding the 10CFR 50.59 evaluation process.

Though not described specifically in this response, Duke Power has completed a preliminary UFSAR review at Oconee and McGuire Nuclear Stations (the Catawba review is in final stages of completion). This review was done in part to address the Nuclear Energy Institute (NEI) initiative (NEI 96-05) and to gather information to determine resources necessary to perform a complete review of each station's UFSAR. Plans for this review will be the subject of separate correspondence with the NRC.

2.0 RESPONSE TO REQUEST (a)

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

2.1 OVERVIEW

The following discussion provides a general overview of the configuration change process at Duke Power Company. Programs and processes are used to control configuration changes, provide engineering design control, achieve the objectives of the Configuration Management Program (which is also described below), and implement 10CFR 50.59, 10CFR 50.71(e), and 10CFR 50, Appendix B.

Figure 2-1 shows the general, simplified process for dealing with configuration changes within Duke Power. Within each diagram box, an identifier is added to point the reader to the paragraph or section which describes the program(s) or process(es) that controls the activity/activities.

Definitions:

Design Bases: Refer to 10CFR 50.2. These design bases are supplemented by and refer to design codes, standards, regulatory requirements and engineering analyses to provide specific design criteria and parameters.

Design Deliverable Documents: Design Output Documents as described by ANSI N45.2.11-1974 (e.g., engineering and test acceptance criteria drawings, certain engineering databases, certain engineering lists, engineering instructions, design specifications, installation specifications, Design Basis Documents) which result from design analyses output and the translation of design bases requirements. These design deliverable documents describe the plant design requirements in a form that can be used for the manufacture, construction, operation, maintenance and/or testing of structures, systems, and components. Certain vendor documents as described in Section 2.2.2.2 are also considered design deliverable documents.

Operational Configuration: The envelope within which the plant is operated and the arrangement of systems and components established by approved operational procedures (normal operating, emergency operating, maintenance, test, surveillance, startup, shutdown, etc.). Operational configuration also includes equipment manipulations for plant operations.

Design Configuration: The arrangement of structures, systems, and components defined by the design deliverable documents.

Configuration Change: The alteration of the plant operational configuration, plant design configuration, or the design bases.

Operational Configuration Change: The alteration of the plant operational configuration.

Design Configuration Change: The alteration of the plant design configuration.

<u>Relationship Between Design Bases/Design Deliverable Documents/ Operational Configuration:</u>

The operational configuration is intended to be maintained within the design provided by the design deliverable documents. The design provided by the design deliverable documents is intended to be maintained within the design bases. This is graphically depicted in Figure 2-2.

<u>Configuration Management Program</u>

The Configuration Management Program (CMP), described in Nuclear System Directive (NSD) 106, is a set of programs and/or processes whose objective is to provide reasonable assurance that the three (3) major components of the CMP:

- the plant design requirements (including the design bases),
- the plant documentation (design analyses/design deliverable documents/UFSAR/databases/lists/procedures, etc.), and
- the as-built plant configuration,

are in agreement. Changes to these components must be approved by authorized personnel. The conformance of these three components is auditable. This is graphically depicted in Figure 2-3.

2.2 PROCESS DESCRIPTIONS

The discussion below is arranged by Operational Configuration Changes and Design Configuration Changes to follow as closely as possible the process chart of Figure 2-1. Some of these processes are used for both operational and design configuration changes and following the process flow exactly would result in duplicate descriptions of the same processes. In these cases of duplication, a reference is provided to the controlling processes.

2.2.1 Processes For Operational Configuration Changes

The operational configuration changes discussed in this Section fall into two categories: equipment manipulations and the installation of replacement items.

Operational configuration includes equipment manipulations required for plant operations and these manipulations are considered in the strict sense to be operational configuration changes. These changes are performed in accordance with approved plant procedures developed as described in Section 2.2.3.6 and supported by the Removal and Restoration Process described in Section 2.2.3.1.

The other operational configuration change is the installation of replacement items as a maintenance activity. The replacement items are evaluated in accordance with the Nuclear Procurement Engineering Program (NPEP) as described in Section 2.2.3.4 to determine if the replacement items are acceptable and if installation as a maintenance activity is acceptable. If so, the replacement activity is performed in accordance with the programs and processes above and the Work Process described in Section 2.2.3.3.

These processes provide reasonable assurance that the operational configuration changes remain within the Technical Specifications and design deliverable documents (see Figure 2-2) which result in plant configuration within previously analyzed and licensed conditions.

2.2.2 Processes For Design Configuration Changes

2.2.2.1 Nuclear Station Modification Program

Design changes, except for Reactor Core Design discussed in Section 2.2.2.3, are controlled by the Nuclear Station Modification (NSM) Program described in Nuclear System Directive (NSD) 301. The NSM Program also links to and/or includes the integration of the processes for engineering design control, 10CFR 50.59 reviews, 10CFR 50.71(e) updates, and other 10CFR 50, Appendix B processes. The NSM Program:

1. Includes the processes for the review, evaluation, and approval of plant design changes (modifications) based on the approved plant licensing bases and the design bases.
2. Identifies the responsibilities of the various individuals/organizations involved in modifications in order to assure proper interface control.
3. Addresses the requirements for small scope modifications (called Minor Modifications), large scope modifications (called NSMs), and Temporary Modifications.

Major elements of the NSM Program which relate to engineering design control, configuration control, 10CFR 50.59, 10CFR 50.71(e), and 10CFR 50, Appendix B are noted and discussed in general terms within this section. These elements are:

1. Scope
2. Design Control Process
3. Technical Specification/Selected Licensee Commitment (SLC)/UFSAR Review and Update
4. Procurement
5. Implementation Requirements
6. 10CFR 50.59 Evaluations
7. Procedure Review and Revision
8. Training Impact
9. Implementation Approval
10. Implementation and Close-out

Each of these elements is discussed below.

1. NSM Program: Scope

   In general, the NSM Program applies to structures, systems, and components that are located within, and are considered part of, the nuclear facility (Reactor Core Design activities are not in the NSM Program scope and are addressed in Section 2.2.2.3). Proposed changes to the plant or changes to design deliverable documents are reviewed for NSM Program scope.

The NSM Program boundary excludes the majority of commercial facility structures and selected support/office areas at the nuclear sites. However, tie-ins to the operational plant systems are performed through the NSM Program.

2. NSM Program: Design Control Process

The design of modifications is subject to the Design Control Process requirements in Section 2.2.2.2. Design analyses are performed or revised, design deliverable documents are revised or developed, and appropriate vendor technical documentation is obtained and reviewed for SSCs affected by the modification. These design deliverable documents are identified as implementation documents and distributed with the intent of assuring that personnel are using technically adequate and correct information.

Errors and deficiencies noted in the design of a modification are corrected by means of a variation notice or a revision to the modification.

The impact of the modification on the In Service Inspection (ISI) and In Service Testing (IST) Programs is also reviewed.

3. NSM Program: Technical Specification/Selected Licensee Commitment (SLC)/UFSAR Review and Update

During the modification design process, the Technical Specifications, Selected Licensee Commitments (SLCs), and the UFSAR are reviewed for possible impact due to the modification. If Technical Specification changes are required in order to implement a modification, the proposed changes are noted and forwarded to the site Regulatory Compliance Group for processing and to obtain the necessary internal and NRC reviews. If Technical Specification changes are required, the modification is not implemented until NRC approval is received.

If SLC or UFSAR changes are required in order to implement the modification, the changes are noted and forwarded to the Regulatory Compliance Group for processing. The UFSAR changes are retained by Regulatory Compliance for incorporation into the next scheduled UFSAR update (see Section 2.2.3.8). Changes to the SLCs and UFSAR are subjected to a 10CFR 50.59 evaluation as discussed in Section 2.2.3.7.

4. NSM Program: Procurement

Procurement needs are reviewed and identified during the modification design process. The NPEP described in Section 2.2.3.4 provides the governing process for any procurement needed during the design or implementation of the modification.

5. NSM Program: Implementation Requirements

As the modification design progresses, the implementation requirements for the modification are developed. Modifications are implemented in accordance with the design deliverable documents previously discussed and the procedures and test plans as noted in the following paragraphs.

Procedures are used during the implementation of modifications to control and coordinate work and testing activities. Criteria are provided to aid in determining if existing station procedures are adequate for the work scope being performed or if the development of a temporary implementation procedure is required. Procedure development and revision are governed by the Procedure Control Process as discussed in Section 2.2.3.6.

A Post Modification Test Plan is developed which includes the testing required, the test acceptance criteria (TAC) to be met, and the approved procedures necessary to perform the testing. Successful testing completion provides reasonable assurance the SSCs affected by the modification perform their intended function and meet their design bases. See the additional discussion of the Testing Program in Section 2.2.3.5.

6. NSM Program: 10CFR 50.59 Evaluations

The final design and the procedures used for implementation of the modification are subjected to a 10CFR 50.59 evaluation. The 10CFR 50.59 evaluation process is discussed in Section 2.2.3.7.

7. NSM Program: Procedure Review and Revision

As part of the final modification package, a list of affected design deliverable documents for the modification, including pertinent vendor documents, is compiled. This list, along with other information about the modification, is forwarded to appropriate plant groups (e.g., operating groups, other engineering groups) for review. This review is performed in order for procedures affected by the modification to be

revised, developed, or removed from use. Operations procedures for modified SSCs are reviewed and revised or removed from use as needed prior to the modified SSC being returned to service. Procedure revision, development, or removal from use is governed by the Procedure Control Process as discussed in Section 2.2.3.6.

8. NSM Program: Training Impact

If the modification significantly alters the function, operating procedure, or operating equipment, then training to station groups is administered as necessary. Modification information is reviewed for impact to the plant simulator, plant simulator training, and Operator Requalification training.

9. NSM Program: Implementation Approval

Plant modifications affecting QA Condition 1 SSCs are not implemented until approvals are received as described in each plant's Technical Specifications. As discussed previously, if Technical Specification changes are required or a USQ is identified, the modification is not implemented until NRC approval is received.

10. NSM Program: Implementation and Close-out

Modifications are implemented through the Work Process (see Section 2.2.3.3) using the information discussed in part 5. NSMs: Implementation Requirements. The as-built condition of the modification is verified. Prior to returning a modified SSC to service, implementation activities, post modification testing activities, and necessary revisions to or removal from use of operations procedures for the modified SSC are verified to be complete. Documents identified by the station staff as Vital To Operations (VTO) are revised to reflect the as built configuration.

Once modified SSCs are returned to service, non VTO documents are revised and distributed to reflect the as built condition of the plant, training is updated as needed, and the Regulatory Compliance group is notified for UFSAR updates or other regulatory commitments. These activities are performed in order for the plant documentation to represent the actual as built plant configuration. Engineering personnel review the design documents completed for the modification for the purpose of assuring appropriate documents have been updated and transmitted.

Identifiable and retrievable documentation of station modifications is retained for the life of the station through microfilming and retention of the modification file and the procedure files.

2.2.2.2 Design Control Process

The design of QA Condition 1 SSCs is controlled for the purpose of assuring that regulatory requirements and the design bases are correctly translated into drawings, specifications, instructions, and procedures.

Computer programs are controlled in accordance with NSD 800 whereby programs are certified to demonstrate their applicability and validity.

Preparation Of Design Documents

QA Condition 1 design documents, such as design analyses and design deliverable documents (e.g., engineering and test acceptance criteria drawings, certain engineering databases, certain engineering lists, engineering instructions, design specifications, installation specifications, Design Basis Documents), are prepared in accordance with approved Engineering Directives Manual (EDM) procedures EDM 101, EDM 130, and EDM 170. The Preparer specifies and includes appropriate codes, standards, UFSAR design criteria and other commitments as applicable, design bases, and other design input (consistent with those design inputs described in ANSI N45.2.11 - 1974) within the design documents. The Preparer notes deviations or changes from such standards within the design documentation package.

Revisions to design analyses are reviewed to confirm that the changes are compatible with the existing design such that the intended function and design bases conformance are still achieved.

Design deliverable documents are prepared/revised:

- to translate/transfer design analysis output,
- to translate/transfer design bases/design criteria information,
- to translate/transfer design analysis output revisions to affected documents,
- to show pending modifications to plant design configuration, and/or

- to show as built plant design configuration after modifications,

in a form that can be used for the manufacture, construction, operation, maintenance, and testing of structures, systems, and components.

Design Verification

Design documents are verified for conformance with applicable codes, standards, UFSAR design criteria and other commitments as applicable, design bases, other design input (as described in ANSI N45.2.11 - 1974 and specified within the design documentation package). Design documents are also verified to confirm that changes are compatible with the existing design, such that the intended function and design bases conformance are achieved.

Design verification consists of design reviews, alternate calculations, and/or qualification testing. The depth of design review is commensurate with the nuclear safety significance of the SSC, complexity of the design, and similarity to previous designs. Design reviews are intended to verify the correctness of design inputs, the translation/transfer of necessary design analysis output to design deliverable documents, logic, calculations, and analyses. Design analyses by alternate methods provide further assurance that, for instance, computer codes are performing as expected, and that no systematic error in calculation procedures exists. Qualification testing, when applicable, is guided by Duke Power's adoption of various regulatory guides related to qualification testing and is reviewed and approved by the responsible engineer.

Design verification is performed in accordance with approved procedures of EDM 101, EDM 130, and EDM 170. These procedures identify the responsibilities, features, and pertinent considerations to be verified such as independence of verifier and approver, verification method, design parameters, acceptance criteria, and documentation requirements. A review of the design document is made with the intent of assuring incorporation of necessary quality assurance information. The design documents are approved by the first line manager/supervisor or above (unless otherwise designated) having responsibility for the design function.

Prior to the release of QA Condition 1 design deliverable documents (e.g., engineering and test acceptance criteria drawings, certain engineering databases, certain engineering lists, engineering instructions, design specifications,

installation specifications, Design Basis Documents), the documents are reviewed to evaluate coordination of other engineering disciplines. If the documents clearly involve no coordination with the other disciplines, this review may be waived by the document sponsor, with documented concurrence by the other disciplines. This discipline coordination provides for design interface within the organization responsible for the design and provides an opportunity for assuring the SSC design is properly integrated among interfacing disciplines and the intended function and design bases are being met.

Vendor documents (e.g., stress reports, design analyses, instruction and maintenance manuals, drawings) are also used for demonstrating QA Condition 1 SSCs perform their intended functions and conform to their design bases. Vendor information is sometimes used as design input to a design document or forms the basis for a design document. For these reasons, vendor documents are part of the design deliverable documents that describe the plant design and those documents pertaining to QA Condition 1 SSCs are reviewed, including coordination of disciplines, approved, and released or transmitted in a manner similar to design documents. The vendor information is subject to these reviews for the purpose of verifying the information meets specified procurement and design requirements and is translated/transferred to design documents which may use the vendor information.

Design verification is required to be completed before relying on the item to perform its function. As part of the design verification completion, necessary supporting and sub-tier documents and documents affected by changes are revised or developed.

Document Control

Document Control is an important part of the Design Control Process. Document Control is governed by numerous implementing procedures but primarily those in EDM 101, EDM 130, and EDM 170; NSD 301, NSD 701, NSD 702, and NSD 703; the Document Management Manual; the Nuclear Procurement Engineering Program (NPEP) Manual; and the Record Retention Schedule.

A master copy of controlled documents (internal documents and vendor documents) applicable to the station's SSCs is maintained in the document control area of each station. These controlled documents are utilized, as appropriate, in the performance of QA Condition 1 activities. Copies of controlled documents are distributed by station document control personnel using a distribution index. These controlled copies provide personnel

with technically adequate and correct information for the purposes intended. The original controlled documents are maintained and controlled in accordance with the directives noted above. Document retention requirements are developed to comply with legal and regulatory requirements.

The flow of design information outside of the Engineering organization is controlled by the transmittal of design deliverable documents, vendor documents, and revisions using distribution indices as described above. Receipt acknowledgment of documents pertaining to QA Condition 1 SSCs is required as specified in the controlling document management procedures. A master file of documents is maintained and a master index, updated regularly, is used to identify documents, revisions, numbers of copies, and distribution. Design documents are maintained, controlled, and revised by Engineering. As design deliverable documents are received from Engineering, superseded copies are required to be destroyed or marked superseded.

Vendor Technical Information

A summary of the control processes for vendor technical information is noted below.

Vendor technical information is defined as updated instruction and maintenance manuals, technical information bulletins, revised technical procedures, and updated replacement part information.

Vendor technical information received as part of equipment procurement during nuclear station modifications is subject to the Nuclear Station Modification (NSM) Program as described in NSD 301. Through the NSM Program, groups responsible for station procedures are notified to review and revise their respective procedures as needed.

Vendor technical information received through the replacement/spare part procurement process that is related to part number changes and/or minor changes in material/design is handled via the Nuclear Procurement Engineering Program.

Vendor technical information received through Vendor Information Letters, Vendor Information Notices, and/or normal business contacts or relationships with equipment suppliers is handled through NSD 204, Operating Experience Program (OEP). If the review of OEP information determines that site vendor documents require revision, a modification is generated to place the activity in the NSM Program. Through the NSM Program, groups responsible for the station procedures are notified to review and revise their respective procedures as needed.

This process provides reasonable assurance that vendor technical information received for QA Condition 1 SSCs becomes incorporated into an existing program that assures the information is complete, accurate, current, and controlled throughout the life of the plant, and appropriately referenced or incorporated into plant procedures, vendor manuals, and design documents.

2.2.2.3 Reactor Core Design

Reactor core design is governed by a program separate from the NSM Program. However, engineering design control, 10CFR 50.59, 10CFR 50.71(e), and 10CFR 50, Appendix B processes are used as described in this response.

Computer programs are controlled in accordance with NSD 800, supplemented by Nuclear Engineering (NE) workplace procedure NE-114, whereby programs are certified to demonstrate their applicability and validity. Major computer programs are submitted and reviewed by the NRC in various Topical Reports.

The reload design process is described in workplace procedure NE-102. Design analyses are prepared, reviewed, and approved for the new core configuration in accordance with the Design Control Process as discussed in Section 2.2.2.2, supplemented by workplace procedure NE-103. The methodology used for these analyses has been approved for use by the NRC via numerous Topical Reports. The output from these analyses is translated into design deliverable documents, checked, reviewed, reviewed across disciplines as needed, and approved in accordance with the Design Control Process discussed in Section 2.2.2.2, supplemented by workplace procedure NE-109. Vendor documents are reviewed and approved in accordance with the Design Control Process previously discussed, supplemented by workplace procedures XSTP-101 and NE-108. Design deliverable documents are distributed in accordance with the Document Control Process previously discussed, supplemented by workplace procedure NE-109.

The design deliverable documents include the parameters that define the acceptable operating limits of the core. The operating limits are defined by the Core Operating Limits Report (COLR). The COLR is revised as necessary based on the design analyses for the core configuration and distributed as described in Section 2.2.3.9 and Compliance Functional Area Manual (CFAM) Directive 3.13.

If a Technical Specification change or unreviewed safety question is involved, a reload report and justification for the Technical Specification change is prepared and submitted to the NRC for

review and approval. Otherwise, a 10CFR 50.59 evaluation is prepared for the reload core and COLR. Revisions to the COLR and the core configuration design are subjected to a 10CFR 50.59 evaluation as discussed in Section 2.2.3.7, supplemented by workplace procedure NE-104.

Technical Specifications are reviewed and necessary changes are processed as discussed previously in the NSM Program and in Section 2.2.3.9. The UFSAR is reviewed and necessary changes are forwarded as discussed in Section 2.2.3.8 on the 10CFR50.71(e) update process.

Procurement is accomplished in accordance with the Procurement Control Process discussed in Section 2.2.3.4, supplemented by workplace procedure NE-111.

Implementation requirements are distributed through design deliverable documents, specifically Engineering Instructions, and include as examples the Final Core Load Map and the Startup and Operations Report.

The core configuration is subjected to the physics testing program described in the UFSAR which defines minimum acceptable testing. The test acceptance criteria are also noted in the UFSAR and are incorporated into site procedures through the Procedure Control Process described in Section 2.2.3.6.

Implementation activities are controlled by procedures as discussed in Section 2.2.3.6.

2.2.3 Generic Processes For Operational Configuration Changes And Design Configuration Changes

2.2.3.1 Removal And Restoration

The Removal and Restoration Process (R & R) is a systematic method the Operations and Chemistry personnel at each site use to remove equipment from service and return that equipment to service. Procedures, directives, checklists, and controlled drawings are used to assure system configuration is maintained during the equipment removal evolution and the return of that equipment to service.

R & R is a commonly used term in the industry that refers to an operational configuration control process. Operating systems are placed in service per operating procedures supporting normal operations or standby emergency availability. When these operating systems are to be removed from service for maintenance or modification, the R & R Process controls how the component or

system is removed from service. Operational reviews are completed with the intent of assuring the removal activities are permitted by Technical Specifications and operating standards. Proper verification activities take place with the intent of assuring the correct components are removed from service. The sequence for removal and restoration steps is also specified when required. Proper authorization, included in the R & R Process, is intended to assure system availability is maintained as required by the Technical Specifications and UFSAR.

The R & R Process also includes restoration activities intended to assure that equipment removed from service is returned to service.

2.2.3.2 Tagout Program

The Tagout Program is part of the R & R process. While the R & R process identifies the components that are to be disabled and tagged, the Tagout Program as described in NSD 500 controls the actual equipment isolation process.

For removal from service activities, the Tagout Program provides directions for implementing identified equipment isolations and guidance for identifying and documenting the isolated equipment. Personnel are expected to not operate or change the isolations to tagged equipment without proper authorization and approvals through the R & R process.

For restoration to service activities, the Tagout Program provides guidelines for removing the equipment isolation, returning the equipment to its proper position, removing the identification, and documenting that the restoration activities have been completed.

2.2.3.3 Work Process

Maintenance activities are completed per the requirements of the Work Process Manual. This collection of directives provides requirements for controlling plant maintenance activities such that appropriate plant configuration is maintained. For normal repair of plant components and periodic service to those components, controls are in place with the intent of assuring replacement parts used have the same fit, form and function. Technician training, maintenance procedures, work practice standards, and plant drawings are used to manage plant configuration during normal maintenance activities.

Plant changes authorized by the NSM Program are implemented using directives in the Work Process Manual. Technicians review NSM packages prepared by Engineering and plan details for implementation of the change with the intent of assuring that only changes described by the NSM package will be made. Periodic checks by QC Inspectors add assurance that the plant change is completed per the drawings and specifications in the design package. These processes are intended to assure that the design change is implemented as specified by Engineering so that there is continuity between actual plant configuration and the drawings representing plant configuration.

Following maintenance and modification activities, appropriate testing is performed with the intent of assuring that components are capable of performing their intended function. Performance of specified post-maintenance and post-modification testing (see Section 2.2.3.5) is required by the Work Process Manual. The work process is intended to assure that maintenance and modification activities and appropriate testing are completed prior to returning the system to service.

The Work Process Manual includes directives that describe when components may be removed from service. Equipment is removed from service according to the requirements of Technical Specifications not to exceed specified durations.

2.2.3.4 Procurement Control Process

NSD 302 and the implementing procedures of the Nuclear Procurement Engineering Program (NPEP) describe the processes related to procurement of items for use at Duke Power Company nuclear stations. The Procurement Control Process requires the control of QA Condition 1 items or services purchased from a supplier, sub-supplier or consultant. This process provides reasonable assurance that:

- quality assurance, technical, design basis, and regulatory requirements are appropriately incorporated into the procurement and vendor documents,

- vendor documents/technical information is reviewed and approved, and

- the procured item is manufactured, delivered, stored, and inspected per procurement requirements.

If a change of an item identified during the procurement process exceeds the defined criteria within an NPEP procedure, then the change is required to be evaluated under the NSM Program. This

is intended to assure design basis issues are reviewed and resolved and 10CFR 50.59 evaluations are performed as appropriate.

Computer programs are controlled in accordance with NSD 800 whereby programs are certified to demonstrate their applicability and validity.

Preparation of Procurement Information

The procurement requirements applicable to each item are determined by a cognizant individual and incorporated as appropriate into the procurement process (e.g., procurement specifications, Commercial Grade evaluations, replacement part evaluations, receipt inspections requirements). The procurement requirements include:

- Quality assurance, technical, design basis, and regulatory requirements appropriate for the item procured.

- Form, fit, and function requirements or references thereto based on the design and safety functions of the item being procured.

- For commercial grade items, critical characteristics determined and approved by engineering personnel based on the manufacturer's published specifications and intended safety function for the items. These critical characteristics are used for acceptance and dedication of commercial grade items.

- For replacement items, technical and quality requirements at least equivalent to those applicable to the original equipment or those specified by a properly reviewed and approved revision.

- Information required from the supplier (e.g., certificate of conformance, vendor analyses, vendor drawings, vendor instruction/maintenance manuals).

- Supplier selection. QA Condition 1 material, equipment and services procured as basic components are only procured from qualified suppliers. Supplier selection for commercial grade items is the responsibility of engineering personnel.

A technical evaluation is performed for QA Condition 1 replacement items and the evaluation is required to address changes in the item from those originally or currently installed. A change is required to be evaluated to reasonably determine that the application and quality requirements of the replacement item are adequate and will not change the safety function of the parent component or system in which it is used. The evaluation results are included in a controlled document which serves as the basis document for configuration control. In many cases, a revision to the specifically affected design document, e.g., manufacturer's drawing, instruction manual, is required by the procurement document. Should this occur, NPEP procedures specify when the NSM Program must be entered.

Procurement information for approved QA Condition 1 materials, parts and components is reviewed for the purpose of assuring that quality assurance, technical and regulatory requirements including supplier documentation requirements are adequately incorporated into the purchase document(s). Significant changes to the content of procurement information are reviewed and procurement documents affected are revised and approved in a manner consistent with the original document.

Procurement Verification

Verification of critical characteristic(s) acceptability for commercial grade items will be by manufacturer/supplier survey, manufacturing surveillance, receipt tests or inspections, or post installation testing. Historical data, when documented, may be used to supplement the other acceptance methods. Dedication will not typically be based purely upon historical data unless such data represent industry wide experience.

Supplier review, audit, and surveillance may be used in order to provide reasonable assurance that materials are in conformance with procurement requirements. Procedures are established which implement the surveillance program for suppliers. This is intended to assure that items and services procured for use in QA Condition 1 applications are in compliance with applicable procurement requirements/specifications.

Suppliers furnish documentation specified in the procurement documents which identifies the material and equipment purchased, the specific procurement requirements met by the items, any procurement requirements which have not been complied with, and a description of any deviations and repair records. This documentation is reviewed, approved, and must be on site with procurement, inspection, and testing requirements satisfied before the item is placed in service. The review of vendor

documents/technical information received is conducted as described in Section 2.2.2.2.

Upon receipt, QA Condition 1 materials, parts and components are required to be placed in a controlled, designated area and subjected to a receipt inspection. This inspection is intended to determine whether or not each item received conforms with applicable procurement requirements. Such inspections and the subsequent determination of conformance or nonconformance are documented by means of reports, which are retained on file. Until a determination of conformance is made, a QA Condition 1 material, part or component is not permitted to be issued and installed except in certain situations where parts or components may be used on a conditional basis. Determination of conformance is made prior to declaring the component or system operable.

2.2.3.5 Testing

Duke Power's nuclear station testing philosophy is documented in NSD 408. The objective of this directive is to provide assurance that SSCs will perform as designed through a program of Periodic (including Predictive Maintenance), Post-Maintenance, and Post-Modification Testing.

Testing is performed at the highest practical system level and as close to design basis conditions as practical to reasonably assure the component and system will perform as designed. Where this philosophy is not practical, an evaluation is performed and documented in controlled documentation (e.g., Problem Investigation Process described in section 5.0, Design Basis Documents, Work Management System, design analysis). Testing is performed by qualified personnel using approved plant procedures to demonstrate that SSCs perform their intended function(s).

Each station is required to develop and maintain testing programs as required by Technical Specifications, SLCs, UFSAR, and other licensee documents. Test acceptance criteria (TAC) are developed for SSCs to verify that Design Basis requirements are met. Testing of QA Condition 1 SSCs is performed by qualified personnel using calibrated instrumentation and acceptance criteria.

The Engineering organization at each site is responsible for the development, maintenance, and revision of TAC drawings as necessary to support changes in plant configuration which affect the SSC's design bases. EDM 130, EDM 170, and NSD 301 provide further information on and the design control requirements for the TAC drawings.

2.2.3.6 Procedure Control Process

Procedure Development

Plant procedures are developed by knowledgeable individuals and follow the requirements for procedures set forth in NSD 703. This directive specifies that procedures meet the requirements set forth in Technical Specifications, SLCs, UFSAR, other licensing documents, NSDs, site directives, Design Basis Documents (DBDs), other design deliverable documents, and applicable vendor documents. This preparation is to assure necessary technical requirements are included in the procedure.

Procedure Review

Prior to approval, each procedure is required to be reviewed by a Qualified Reviewer with the objective of verifying that information contained within the procedure:

- is accurate, complete, and that sufficient documentation is contained to assure that the intent of the procedure is fulfilled,
- satisfies the requirements set forth in the Technical Specifications, SLCs, UFSAR, other licensing documents, NSDs, site directives, DBDs, other design documents, and vendor documents, and
- does not place the facility in violation of Technical Specifications or Selected Licensee Commitments, or in operation outside the previously analyzed design shown on design deliverable documents.

In addition, the Qualified Reviewer specifies any needed cross-disciplinary review. These reviews provide further scrutiny of design bases and activity implementation requirements. Additional reviews by a Qualified QA Reviewer are required for certain categories of procedures.

Procedures are subjected to a 10CFR 50.59 Evaluation as described in Section 2.2.3.7.

Procedure Revisions

Procedure revisions are processed in the same manner as a new procedure with the exception that revisions classified as minor changes (e.g., editorial corrections, non-technical changes) may be processed without the performance of a 10CFR 50.59 Evaluation.

Procedure Approval

Procedure approval is performed as required by Technical Specifications.

2.2.3.7 10CFR 50.59 Evaluation Process

10CFR 50.59 evaluations are performed following the guidelines in NSD 209. This directive incorporates the guidance provided in NSAC-125, "Guidelines for 10CFR 50.59 Safety Evaluations." Additional guidance has been provided to individuals involved in the 10CFR 50.59 Evaluation process to reflect the guidance in NEI document NEI 96-07, Draft A1, "Guidelines for Performing 10CFR 50.59 Safety Evaluations" and reinforce the requirements of the 10CFR 50.59 regulation.

The need for performing a 10CFR 50.59 evaluation is described in the program or process used for an activity (e.g., final modification design, procedure development/revision, UFSAR changes). The evaluation is a two step process. The first step is a screening of the proposed activity to determine if the 10CFR 50.59 regulation is applicable. If Technical Specification changes are required for the activity, the activity cannot be reviewed under 10CFR 50.59 and will be processed under the applicable regulations. If changes to programs (e.g., QA Program, Security Plan, Emergency Plan, or their implementing procedures) which are covered under other regulations are required, the changes are reviewed against the requirements in the appropriate regulations to provide reasonable assurance changes are not implemented until NRC approvals are received, if necessary.

The second step is performed once the 10CFR 50.59 regulation is determined to be applicable. A safety evaluation is conducted which provides the basis for determining whether the activity involves an unreviewed safety question (USQ). The safety evaluation addresses the questions described in 10CFR 50.59 and provides a discussion and justification for the answers to those questions based on the information known about the proposed activity. The 10CFR 50.59 evaluation is reviewed by an individual other than the individual performing the evaluation.

This evaluation and the review thereof are documented and placed in the appropriate file for retention purposes.

If a USQ is identified during the safety evaluation, the activity may be revised to eliminate the USQ. Otherwise, the proposed activity is internally reviewed and evaluated by the applicable site Plant Operations Review Committee (PORC) and the Nuclear Safety Review Board (NSRB), Duke's independent safety review committee, prior to submission to the NRC for approval of the activity. The activity is not performed/implemented until NRC approval is received.

Technical Specification changes as mentioned above are internally reviewed and approved by the site PORC and the NSRB prior to submission to the NRC for approval of the change. The activity is not performed/implemented until NRC approval is received.

Conformance to the criteria in 10CFR 50.59 provides reasonable assurance that changes to the plant configuration are controlled such that the plant remains within its licensing and design bases as described in the UFSAR and other licensing documents.

2.2.3.8 10CFR 50.71(e) UFSAR Update Process

Duke Power has written guidelines for revising the Updated Final Safety Analysis Report (UFSAR). These guidelines for the preparation and NRC submittal of revisions to the UFSAR are intended to assure compliance with the requirements in 10CFR 50.71(e). This guidance is contained in Compliance Functional Area Manual (CFAM) Directive 3.7. This guidance has been upgraded and incorporated into a new Nuclear System Directive, NSD 220 (UFSAR Revision Process), to recognize the broader applicability of the UFSAR revision process to groups outside of the Regulatory Compliance Group. This is scheduled to be implemented in 1997.

The need to consider UFSAR revision is described in a number of NSDs since UFSAR changes are normally made through other plant processes. These include NSD 301, Nuclear Station Modifications; NSD 209, 10CFR 50.59 Evaluations; NSD 703, Administrative Instructions for Station Procedures; and those NSDs that make up the corrective action program. The Problem Investigation Process (PIP) Program, (NSD 208), is used to identify, track, and resolve discrepancies found in the UFSAR. The resolution of these PIP items may or may not result in UFSAR revisions, since PIP resolution can also require a plant modification or procedure revision.

NSD 220 reinforces the NSM Program directive, NSD 301, in that it provides for the review of design change packages by site Regulatory Compliance to identify potentially impacted sections of the UFSAR. NSD 220 is also intended to assure that proposed revisions to the UFSAR are compatible with existing licensing commitments or that commitment changes are pursued, as appropriate.

2.2.3.9 Facility Operating License/Technical Specifications

Administrative controls are in place that are intended to assure the control of each nuclear unit's Facility Operating License and Technical Specifications (FOL/TS). CFAM Directive 3.3 (Facility Operating License/Technical Specification) is currently approved and in use to control the development, review, approval, implementation, and documentation of FOL/TS amendments. Currently, CFAM 3.3 is being revised to convert this document to a new NSD. This conversion is intended to enhance control over the FOL/TS amendment process since it will apply directly to a wider segment of the Duke nuclear organization.

The new NSD (NSD 221) is currently undergoing review and incorporates changes based upon experience obtained during the previous use of CFAM 3.3 over a period of more than two years. These changes should strengthen the overall FOL/TS process since corrective actions for previous process problems are included. NSD 221 delineates the responsibilities of personnel involved with the FOL/TS process and focuses on compliance with applicable NRC regulations.

NSD 221 contains the procedural steps necessary to prepare, submit, and implement an amendment to a site's FOL/TS. NSD 221 also applies to each unit's Core Operating Limits Report and Independent Spent Fuel Storage Installation (currently applicable only to Oconee). When a determination is made by a responsible individual that an amendment request is warranted, the NSD outlines steps intended to assure that the final submittal is correct and that areas of concern (such as resultant effects on station procedures, the UFSAR, other outstanding Technical Specification changes, preservation of design basis, and other Duke nuclear stations) are addressed.

NSD 221 places emphasis on the review and approval aspects of the FOL/TS process. Responsible individuals assure that appropriate technical reviews are completed prior to Station Manager approval. Proposed changes are reviewed for technical accuracy, completeness, clarity, impact on the respective areas of responsibility, and impact on other Technical Specifications. For proposed amendments to the nuclear sites' FOL/TS, the final

phases of the review process include formal presentation to, and review by, the applicable site PORC and the NSRB.

Proposed amendments to the FOL/TS are submitted to the NRC pursuant to 10CFR 50.90 and 10CFR 50.92. These regulations require the development of a detailed description and technical justification of the proposed changes and the performance of a No Significant Hazards Consideration Analysis. These requirements include the performance of detailed technical reviews and help further assure that the licensing basis of the facility is preserved. Finally, the proposed FOL/TS amendment is submitted to the NRC for review and approval.

As part of developing the FOL/TS change, affected processes and documents (e.g., plant procedures, DBDs, other design documents) are identified. Once NRC approval of the FOL/TS change is received, the problem investigation process is entered to track needed changes to affected processes and documents.

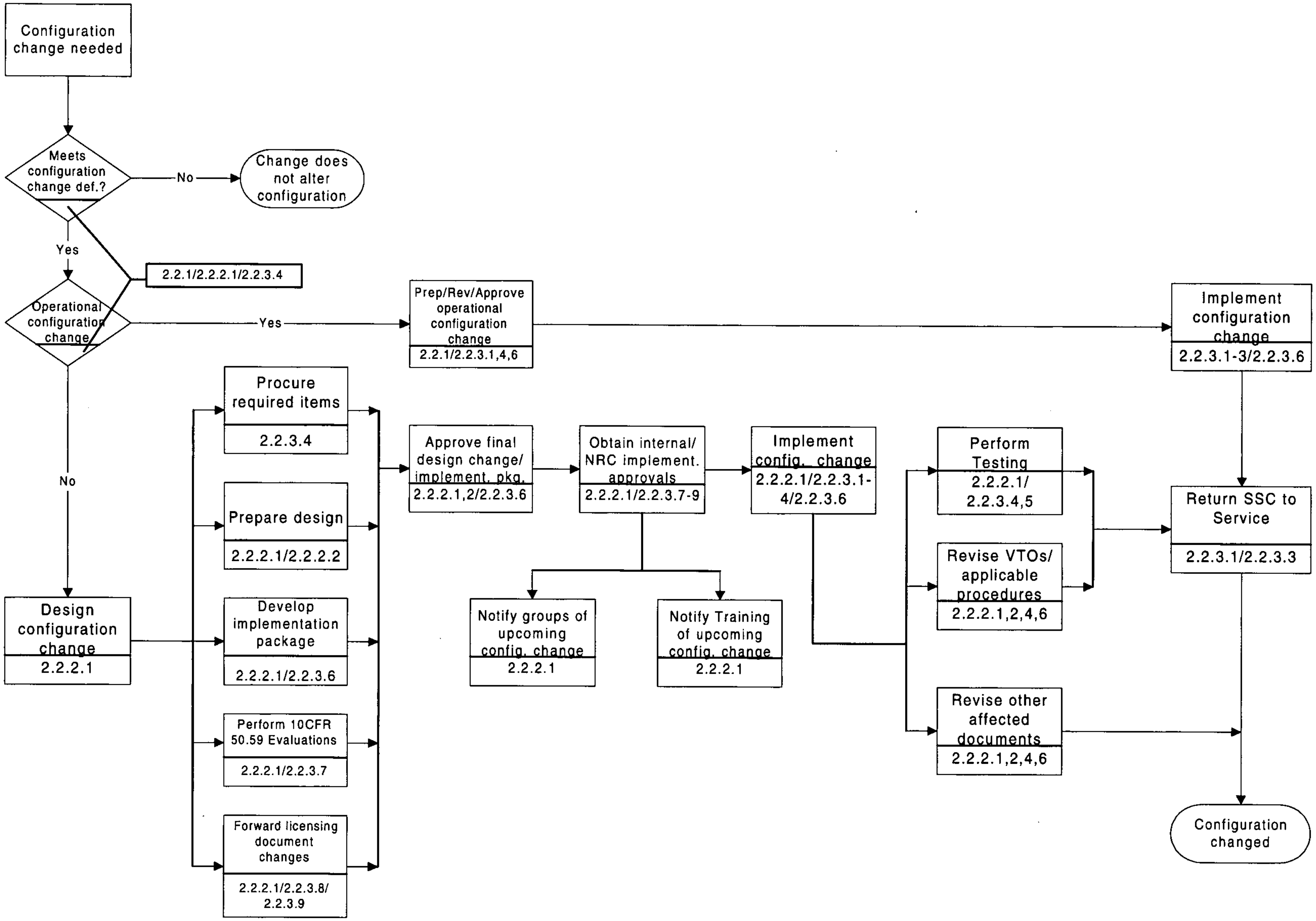

Figure 2-1 Configuration Change Process (Simplified)

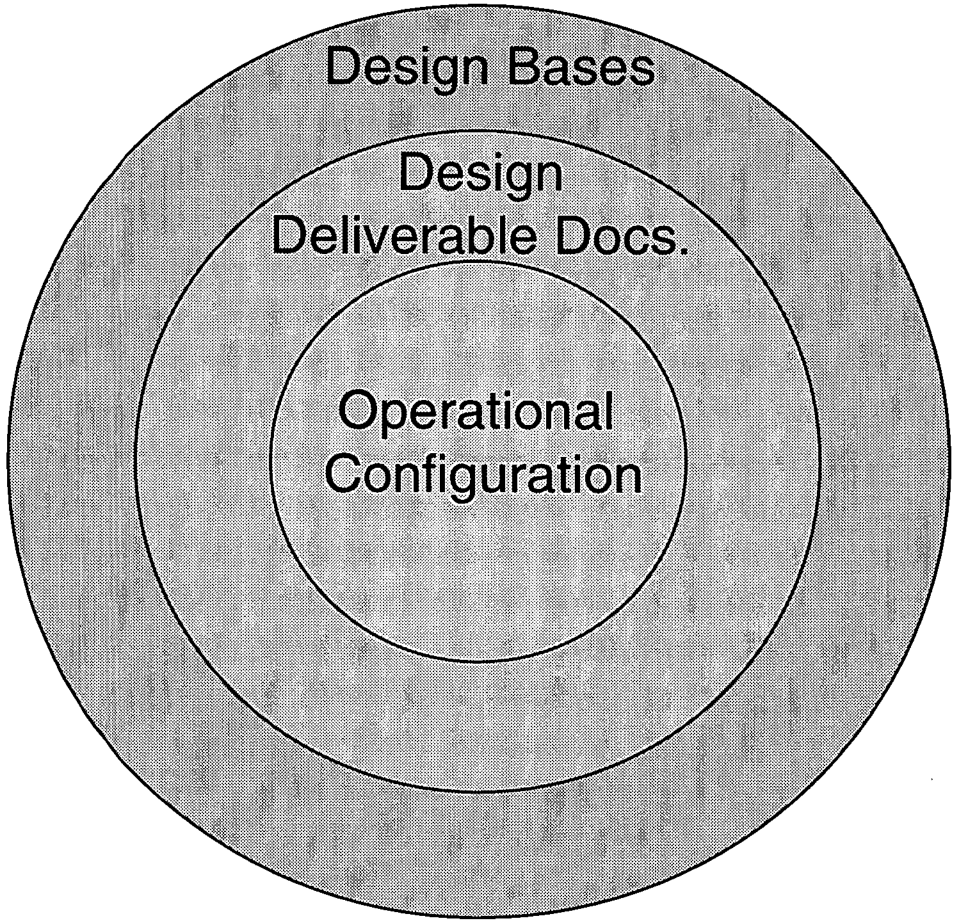

Figure 2-2: Relationship Between Design Bases/Design Deliverable Documents/ Operational Configuration (Ref. Section 2.1)

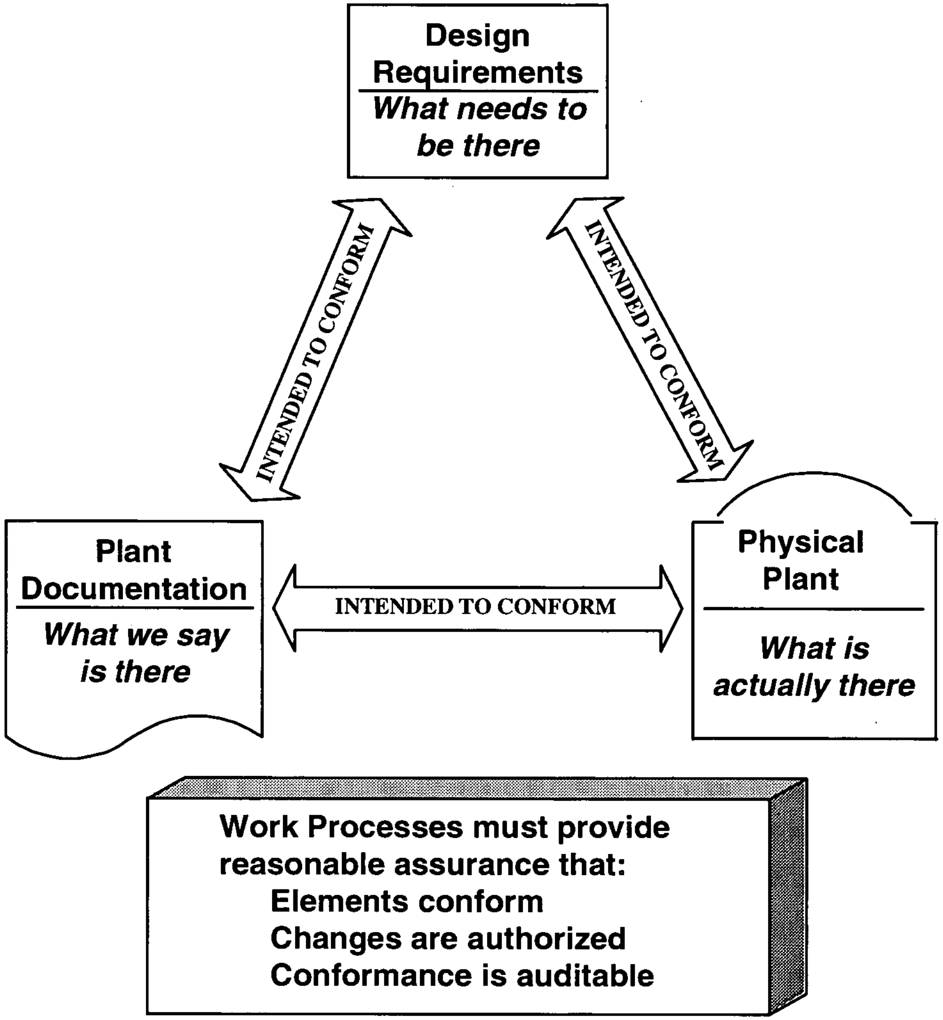

Figure 2-3: Configuration Management Objectives (Ref. Section 2.1)

3.0 RESPONSE TO REQUEST (b)

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

3.1 OVERVIEW

Duke Power Company believes that design bases requirements for the Oconee, McGuire, and Catawba Nuclear Stations have been accurately translated into the appropriate facility operating, maintenance, and testing procedures. Duke's rationale for this belief is based upon a number of factors.

First, Duke relies upon the assumption that when procedures for operations, surveillance testing, and maintenance were originally developed for Oconee, McGuire, and Catawba, those procedures generally complied with the respective design requirements for those plants. This reliance is based upon established processes (discussed in Section 2.0) that have instilled in them certain checks, reviews, and approvals that are in place to develop that level of assurance.

Second, the improved availability of design basis information also supports Duke's rationale. The performance of a Design Basis Review program, begun in 1989 and concluded in December, 1995 at all three Duke nuclear plants has led to the consolidation of design basis information into Design Basis Documents (DBDs).

Third, design and document control processes provide confidence in the proper translation of design bases requirements into procedures. Duke's Design and Document Control Processes are implemented by a number of procedures which collectively limit and control the revision of, and the dissemination of, design information.

Fourth, the procedure control process is another important factor since it provides the mechanism for combining available design, licensing, and other regulatory requirements into implementing procedures for operating, maintenance, and testing activities. Since the initial development of operating, maintenance, and testing procedures for Duke's nuclear plants, controls have been in place for changing those procedures. These change control processes contain the checks and balances necessary to reasonably assure continuing consistency between the design requirements and plant procedures.

Finally, oversight programs also serve to confirm the effectiveness of existing programs in maintaining consistency between design bases and plant procedures.

3.2 AVAILABILITY OF DESIGN BASIS INFORMATION

Duke Power believes that the availability of design basis information is paramount in assuring that it is properly translated into operating, maintenance, and testing procedures. To this end, Duke Power has developed Design Basis Documents (DBDs) which consolidate, and make readily available in an easily retrievable format, design basis information on SSCs (see Appendix A for further discussion). To date approximately 259 DBDs have been written; 90 for Oconee, 83 for McGuire, and 86 for Catawba. These DBDs supplement the design information already available in other design deliverable documents through the normal document control process. DBDs are updated as design changes are implemented using the Design Control, Document Control, and NSM Program processes. This is intended to assure that the DBDs accurately reflect the current plant configuration. The DBDs are readily available electronically through individual workstations to virtually all plant personnel and are also available in controlled hard copy format through each site's Master File.

These DBDs and other design documents are used in the development of plant procedures, which is another part of the rationale that design bases requirements are adequately reflected in plant procedures.

3.3 DESIGN AND DOCUMENT CONTROL

Duke's Design and Document Control processes provide confidence in the proper translation of design bases into design deliverable documents and proper distribution of those documents. These design deliverable documents are used as input into the development and revision of procedures as described in Section 2.2.3.6. The specifics of the Design and Document Control Processes are described in Section 2.2.2.2.

These processes provide reasonable assurance that the design information is complete, accurate, current, controlled throughout the life of the plant, and available for incorporation into plant procedures.

3.4 PROCEDURE CONTROL PROCESS

Procedure control processes are also important in the conclusion that design bases requirements are properly translated into appropriate procedures (discussed in Section 2.0). Plant procedures are developed by knowledgeable individuals and follow the requirements for procedures set forth in NSD 703. This directive specifies that procedures meet the requirements set forth in Technical Specifications, SLCs, UFSAR, other licensing documents, NSDs, site directives, DBDs, other design deliverable documents, and applicable vendor documents.

Prior to approval for use, each procedure is required to be reviewed (see Section 2.2.3.6) in order to verify that necessary information is contained within the procedure.

Procedure revisions are processed in the same manner as a new procedure with the exception that revisions classified as minor changes (e.g., editorial corrections, non-technical changes) may be processed without a 10CFR 50.59 Evaluation.

3.5 INTERNAL OVERSIGHT PROGRAMS

In addition to the rationale presented above, Duke Power's oversight programs also serve to confirm the effectiveness of the various programs and processes in maintaining consistency between design bases and plant procedures.

Included in Duke Power's oversight program is a program of self assessments, regulatory audits and technical audits (discussed in Section 6.0). These oversight programs include, but are not limited to, self assessments (i.e., in-plant reviews, audits, and performance assessments) and high level oversight independent review committee activities such as the Plant Operations Review Committee (PORC) and the Nuclear Safety Review Board (NSRB). Experience has shown that these oversight programs provide an adequate introspective look at how well the various processes and programs are working. In particular, Duke's Self Initiated Technical Audit (SITA) process is a vertical slice inspection process which has been in place for ten years.

These oversight programs have and will continue to raise issues relating to system or component design bases. When issues are raised, the corrective action program provides the mechanism whereby the issues are identified, tracked, reported and resolved, as necessary.

3.6 RESULTS

Duke's established process controls for procedure development, design and document control, and the availability of design basis information, along with the results of the internal oversight programs that have evaluated the effectiveness and implementation of design basis documents, design control, and operating, maintenance, and testing procedures, have provided reasonable assurance that the design basis requirements have been appropriately translated into operating, maintenance, and testing procedures.

4.0 RESPONSE TO REQUEST (c)

Rationale for concluding that systems, structure, and component configuration and performance are consistent with the design bases;

4.1 OVERVIEW

Duke Power Company's rationale for concluding that systems, structure, and component configuration and performance are consistent with the design bases of the Duke nuclear stations is based on several factors.

First, Duke was in the unusual situation of acting as its own architect-engineer and constructor. As a result, Duke maintained direct control of the processes for assuring that design basis requirements were accurately translated into the configuration and operating limits of SSCs. Second, as were all reactor licensees, Duke was subjected to the NRC's close scrutiny during construction.

Third, each of Duke's nuclear stations underwent a startup testing program before beginning commercial operation. That startup testing provided additional corroboration for the original NRC licensing conclusion that reasonable assurance existed that SSC configuration and performance at each station were consistent with the respective design bases for the stations.

Fourth, since each station began commercial operation, SSC configuration and performance within the applicable design bases has been maintained through routine operation, maintenance and modification activities, and the implementation of controlled procedures and processes. In particular, these procedures and processes (which are described above in Sections 2.0 and 3.0) control design changes and operation, maintenance and testing activities. These processes and procedures also include provisions for the use of design bases information. In this regard, Duke Power has conducted a Design Basis Review program at all three nuclear stations, resulting in consolidation and improved availability of design bases information for a number of systems and design attributes.

Fifth, plant configuration and performance are monitored through normal surveillance, testing, and walkdown activities. Performance testing, for example, provides an additional measure of assurance that SSC configuration and performance remains consistent with design bases.

Finally, Duke Power also relies upon more formal evaluations, such as audits and assessments, and other oversight activities to provide ongoing assurance that the processes which maintain plant configuration and assure plant conformance with design bases continue to meet expectations. In particular, SITAs are performed to look into the various aspects of a system's compliance with its design basis.

Each of these aspects of Duke Power Company's "rationale" in response to information request (c) is discussed in this section.

4.2 INITIAL LICENSING AND DESIGN ACTIVITIES

The fact that Duke Power has always served as its own architect-engineer and constructor has enabled Duke to better assure that design and construction activities are well coordinated. This provided better opportunities for design engineers to closely monitor the proper translation of the system/component design bases through the design process and to eventual construction/installation activities. As NRC inspection of design and construction activities identified non-conformances, resolution could be handled through direct interaction with the design authority (i.e., Duke Power) rather than having to interface with a licensee who would then have to involve a third party architect-engineer in the process. This same close working relationship also existed in the resolution of design issues during the initial licensing activities for all three Duke nuclear stations. Although this is an intangible factor, Duke believes that it is nonetheless one factor that tends to give confidence in the baseline configuration of each station.

Similarly, the startup test program was developed and conducted by Duke Power Company. This program was reviewed by the NRC as was the conduct of the testing. This testing provided a solid baseline for providing assurance that the three Duke nuclear stations configuration and performance was consistent with their design bases.

4.3 AVAILABILITY OF DESIGN BASIS INFORMATION

Duke Power believes that the availability of design basis information is also important in assuring that the configuration and performance is consistent with design bases. To this end, Duke Power has developed Design Basis Documents (DBDs) which consolidate, and make readily available in an easily retrievable format, design basis information on SSCs (see Appendix A for further discussion). To date approximately 259 DBDs have been developed; 90 for Oconee, 83 for McGuire and 86 for Catawba.

These DBDs supplement the design information already available in other design deliverable documents through the normal document control process. DBDs are updated as design changes are implemented using the Design Control, Document Control, and NSM Program processes. This is intended to assure that the DBDs accurately reflect the current plant configuration. The DBDs are readily available electronically through individual workstations to virtually all plant personnel and are also available in controlled hard copy format through each site's Master File.

These DBDs and other design documents are used in the development of plant procedures, which is another part of the rationale that design bases requirements are adequately reflected in plant procedures.

4.4 TESTING AND MONITORING ACTIVITIES

4.4.1 Walkdowns

The configuration and performance of SSCs is monitored and assessed daily by routine plant activities. These include Operations rounds and equipment walkdowns. In addition, system engineers regularly monitor performance of their systems through discussion with plant operators and system walkdowns. When a problem is noted, as a result of testing or monitoring, immediate action may be taken by a plant operator to restore proper system/component parameters (e.g., operator adjusts flow, pressure, temperature). For situations which cannot be corrected immediately, the problem is either entered into the Work Management System as discussed in Section 5.2.1 or entered into the Problem Investigation Process (PIP) program for more detailed evaluations and corrective action, as appropriate.

4.4.2 Performance Testing

Routine surveillance tests of equipment are directed at the performance of systems, structures, and components. These tests include performance testing conducted by both the Operations and the Maintenance departments. Performance testing is conducted at the appropriate system level and as close to design basis conditions as practical to assure the component and system will perform as designed. A comprehensive program of Periodic, Post-Maintenance, and Post-Modification Testing is described in some detail in Section 2.2.3.5. Testing is performed by qualified personnel using approved plant procedures to demonstrate that SSCs meet acceptance criteria.

Each station has ongoing testing programs as required by Technical Specifications, SLCs, UFSAR, and other licensee documents. Test acceptance criteria (TAC) included in DBDs are developed for SSCs to verify that Design Basis requirements are met. Examples of the various testing processes are:

- Integrated Emergency Systems Test
- Full Flow Systems Test
- Surveillance Tests
- IWP/IWV Component Tests

The Engineering organization at each site is responsible for the development, maintenance, and revision of TAC drawings as necessary to support any changes in plant configuration which affect the SSC,s design bases.

Test data is also used to benchmark engineering models. This testing provides confidence in the engineering predictions used to verify certain design features of components.

4.5 INTERNAL OVERSIGHT PROGRAMS

4.5.1 Overview

Duke's oversight programs are intended to provide assurance that programs and processes are effectively implemented and meet applicable requirements. Included in Duke Power's oversight program is an aggressive program of self assessments, regulatory audits and technical audits. These oversight programs include, but are not limited to, self assessments (i.e., in-plant reviews, audits, and performance assessments), and high level oversight independent review committee activities such as Plant Operations Review Committee (PORC) and Nuclear Safety Review Board (NSRB). These oversight programs provide the requisite introspective look at how well the various processes and programs are working.

4.5.2 Self Initiated Technical Audits (SITAs)

Duke's SITA program is a specialized audit conducted to assess the operational readiness and functionality of a selected SSC at a nuclear station using the guidelines detailed in NSD 600, Self Initiated Technical Audits. Particular attention is directed to the details of plant/system modification, design control, operations, maintenance, testing, and quality assurance. These vertical slice audits are similar in approach and thoroughness to NRC safety system functional team inspections.

These audits are conducted at an approximate frequency of every eighteen months for each nuclear site. Audit scope is established by engineering management from all Duke nuclear sites and support locations. Candidate systems for selection are generally based on risk significance, system complexity, and experience during the last one to two years, although other considerations may be applied.

The SITA process examines the particular system in detail and is intended to identify design basis issues, UFSAR discrepancies, documentation errors, procedural inadequacies and other weaknesses or shortcomings in system design or operation. This critical look at important systems has identified a number of issues which have resulted in improvements in documentation and plant design changes. As a result, plant safety has been enhanced and improved understanding of the systems' design bases has resulted. The SITA process is further described in Section 6.5.2 and Appendix B.

4.6 RESULTS

The combination of the original startup testing program, the process controls associated with plant configuration changes, and ongoing testing programs, provide ongoing confirmation that component configuration is maintained and performance is consistent with design bases. Duke's internal oversight programs and, to a lesser extent, NRC inspections (e.g., SSFI at Oconee), have evaluated the effectiveness and implementation of design basis documents, design control, and operating, maintenance, and testing procedures. The results of these evaluations, together with the corrective action processes described in Section 5.0, provide reasonable assurance that SSC configuration and performance are consistent with their design bases.

5.0 RESPONSE TO REQUEST (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;

5.1 OVERVIEW

The processes described throughout this response reflect process requirements, philosophy, and intent. Where departures from the processes are identified, the problem investigation process as described in this section is used to develop appropriate corrective actions.

This section describes the processes used by all Duke Power Company nuclear plants to identify problems, determine the extent and root causes of the problems identified, report problems to the NRC, and resolve problems identified through appropriate corrective actions, including actions to prevent recurrence.

Duke Power's processes for the identification of problems and implementation of corrective actions, which include determination of the extent of problems, actions to prevent recurrence, and reporting to NRC are outlined in various Nuclear System Directives (NSDs) which are applicable to all three Duke nuclear stations. The directives which address these processes are:

NSD 210 - Corrective Action Program
NSD 204 - Operating Experience Program
NSD 208 - Problem Investigation Process
NSD 212 - Cause Analysis
NSD 203 - Operability
NSD 216 - Significant Event Investigation Teams
NSD 202 - Reportability

Collectively, these systems directives, which are followed at all three Duke plants, require site personnel to identify conditions adverse to quality, including nonconformances in design documents, as-built plant drawings, and procedures. They provide a consistent method for identifying problems, reporting problems, investigating such problems through a controlled process, controlling and tracking corrective actions, and performing periodic audits to verify compliance. Requirements are also in effect for trending of specific issues to address programmatic concerns and identify improvements where necessary to preclude recurrence.

Identified conditions adverse to quality are assessed to determine their impact upon operability in accordance with NSD 203. After they are evaluated, conditions adverse to quality are assigned a priority for timely corrective action based upon their significance to safety, and are subjected to root cause evaluations. Problems are tracked through completion of corrective action and are trended to identify potentially significant adverse conditions. Station processes are also in place to prevent recurrence of problems, and to identify and resolve generic issues.

Station processes require that identified conditions adverse to quality be evaluated for reportability to the NRC and, if appropriate, reported to the NRC pursuant to NRC regulations set forth in 10CFR Sections 50.72, 50.73, 50.9, Part 21, and station Technical Specifications.

Responsible plant personnel are trained in the above procedures.

5.2 PROBLEM IDENTIFICATION

5.2.1 Overview

NSD 210 (Corrective Action Program) provides the overall direction and identifies the various input programs (e.g., in-house events, industry operating experience, NRC, Regulatory Audits, NSRB, Security Events, Station Modifications) that constitute the Nuclear Generation Department corrective action program. The problem identification process is intended to capture issues at Duke that are adverse to quality, situations or conditions not in accordance with requirements/expectations, and potential problems external to Duke which need to be evaluated and corrected if applicable to Duke nuclear plants.

Duke's corrective action program is designed to encompass a broad scope of issues. Issues which are captured by the corrective action program include plant problems, design basis issues, drawing problems, audit and assessment findings, Nuclear Safety Review Board concerns, UFSAR discrepancies, NRC violations, INPO findings, and security, insurance, environmental, industrial safety and station enhancement issues.

Moreover, the threshold for entering issues into the corrective action program is low. This low threshold is designed to facilitate the identification and correction of lower level problems before they become significant.

Identified problems are resolved using one of two major processes, the Problem Investigation Process (PIP) and/or the Work Management System (WMS). WMS is designed to identify and resolve routine maintenance problems. As such, WMS is considered as part of the Work Process described in Section 2.0 of this response. By contrast, the Problem Investigation Process is designed to address non-routine problems which require further evaluation. PIP is the primary mechanism by which problems are addressed.

5.2.2 Problem Investigation Process (PIP)

The PIP program, described in NSD 208, is the systematic process whereby a problem is identified, screened for significance, and evaluated as required to determine SSC operability, NRC reportability, cause and appropriate resolution. These steps are described in more detail below. The PIP program is designed such that anyone working at a Duke nuclear plant can identify a problem and generate a PIP.

Duke management expects employees to initiate a PIP when equipment and personnel do not perform as expected, when unexpected changes occur, or when conditions are identified that are inconsistent with requirements or regulations. Computer access and/or PIP forms are provided at a number of locations throughout the nuclear sites for help with PIP initiation by employees.

5.2.3 Other Problem Identification Mechanisms

Another mechanism for problem identification is the Operating Experience Program (OEP), described in NSD 204. The OEP provides for the initial evaluation of external operating experience information and subsequent entry of the information into the Duke corrective action program, if applicable, to any of the Duke nuclear stations. Additionally, the OEP program provides for review of significant problems occurring at one Duke nuclear station for applicability to either or both of the other two Duke nuclear stations.

Industry operating experience is evaluated and documented in the Operating Experience Database. This database is accessible to nuclear generation department personnel. The database has provided a valuable resource to aid in the resolution of significant issues.

Duke internal operating experience for significant events is evaluated and documented by the appropriate PIP in the Generic Applicability section. Operating Experience issues applicable to Duke nuclear plants are processed via the Problem Investigation Process at the affected site(s).

5.2.4 Employee Concerns Program

Duke Power's Internal Audit Department administers an employee report line where any employee can anonymously identify any problem or issue of concern. This entire process is maintained by an independent third party to help shield the individual. Appropriate issues are identified to management and processed by the applicable management policy or process (including but not limited to PIP).

5.3 PROBLEM EVALUATION

5.3.1 Problem Classification

After entry into the PIP program, the newly generated PIP is screened for significance by a centralized screening team. This team is composed of several disciplines with appropriate training in identifying and classifying various problems.

A PIP is classified as either a More Significant Event (MSE), level 1 or 2, or a Less Significant Event (LSE), level 3 or 4 based on the Significance Criteria listed in NSD 208.

5.3.2 Operability Determinations

As part of the problem evaluation process, issues which require a determination of SSC operability are evaluated using the guidelines contained in NSD 203. This directive uses the definition of "Operable" contained in each station's Technical Specifications and the definition of Design Bases contained in 10CFR 50.2. NRC Generic Letter 91-18 was used as a guidance document in the development of this directive. Operability evaluations are documented by or referenced in the PIP program.

5.4 CAUSE ANALYSIS

The cause analysis process is outlined in NSD 212, which provides information on how to conduct a cause analysis using several industry accepted methodologies. This systematic approach provides reasonable assurance that problems are appropriately prioritized, evaluated to determine their extent, and resolved commensurate with the level of significance of the issues.

Depending on the significance of the problem, a root cause analysis or an apparent cause analysis may be performed as specified in NSD 208, Appendix E, PIP Action Category. The type of cause analysis performed is commensurate with the significance of the problem identified. Depending on the significance of the problem, the PIP may require (1) no further action (trending only), (2) an apparent cause evaluation intended to fix the immediate problem and prevent recurrence of the problem, or (3) a root cause analysis to assist in developing corrective actions to prevent recurrence.

Responsibility for performing a Root Cause Analysis is coordinated by the Root Cause Coordinator and performed by one or more Root Cause Evaluators.

5.5 CORRECTIVE ACTION

Upon completion of a proposed resolution, any corrective actions that are developed are assigned to appropriate organizations. The PIP program requires that for any corrective actions identified, both 'Proposed Corrective Action' and 'Actual Corrective Action' be specified and tracked. Any problem can have multiple corrective actions assigned. This allows both immediate corrective actions and long term corrective actions to be specified and tracked independently. Proposed resolution, proposed corrective action, and actual corrective action all require approval by assigned personnel, with overall approval of the issue or concern after completion of all steps. In some cases, PIP items are closed and actual completion of corrective actions is managed in action tracking programs.

5.6 SIGNIFICANT EVENT INVESTIGATION

Plant events determined to be significant by management are investigated and reported on using the guidelines in NSD 216, Significant Event Investigation Teams (SEIT). The SEIT report includes the root cause of the event and recommendations to prevent recurrence. These recommendations are entered into the PIP program and tracked in the same manner as other identified problems. Where appropriate, recommendations are evaluated for applicability to the other Duke nuclear stations.

5.7 NRC REPORTING

As part of the problem identification process, identified problems and issues are evaluated to determine whether NRC notification is required. NSD 202 provides guidance on meeting NRC non-emergency reporting requirements contained in 10CFR 50.72, 10CFR 50.73, 10CFR 21 and 10CFR 50.9. NSD 202 reflects Duke's reporting philosophy, and is intended to foster consistency in reportability decisions among all Duke nuclear stations. Specific examples are provided to help with reportability decisions. Examples are included for both reportable and non-reportable events. Guidance from NUREG 1022 (Event Reporting Guidelines) is included in NSD 202.

5.8 TRENDING OF PROBLEMS

The PIP Program, the NSM Program, the Preventive Maintenance Living Program described in NSD 411, the Maintenance Rule Program described in EDM 210, and the Failure Analysis and Trending (FAT) Program described in EDM 215 all have guidelines for identifying, resolving and correcting trends on less significant or performance related problems. These programs allow emerging, recurring process and performance issues to be identified, tracked and resolved.

5.9 SITA PROCESS AND CORRECTIVE ACTION

Duke has long recognized the importance of conformance with the design bases and has acted on that recognition, in part by developing and performing SITAs. Part of this program interfaces with the corrective action process, to the extent that SITAs are designed to find problems (including deviations from the design bases), determine the extent of the deviations (both at the site at which they are discovered and at other sites), and provide for the determinations of the root causes of significant problems.

Moreover, through the SITA program and the corrective action program Duke identifies and implements comprehensive corrective actions. Corrective actions not only remove significant discrepancies from the design bases (therefore increasing confidence in the consistency today between the plants and their design bases), but also prevent recurrence of similar deviations, thereby increasing confidence in the continuing consistency between the plants and their design bases.

6.0 RESPONSE TO REQUEST (e)

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

6.1 OVERVIEW

Duke Power believes that the processes and programs, in conjunction with the self assessment program described in this response, are effective in providing reasonable assurance that Oconee, McGuire, and Catawba Nuclear Stations are operated and maintained in accordance with their respective design bases. This belief is predicated on a number of factors. First, management ownership of the processes and programs through the Business Excellence Steering Team (BEST) concept provides the appropriate level of management involvement to maintain and improve processes and programs. Second, employees are encouraged to demonstrate a questioning attitude which is intended to capture additional problems and issues that may not be identified through other mechanisms. Third, the corrective action program and, in particular, the PIP program has proven to be an effective tool for capturing a variety of problems, including design basis questions, and assuring that appropriate corrective action and reporting is accomplished. Finally, Duke's self assessment programs, particularly Regulatory Audits and SITAs, provide ongoing assurance that processes and programs are effective. The SITA is the detailed vertical slice audit of a selected system at each nuclear station on a periodic basis.

Each of these points is discussed in the ensuing sections.

6.2 MANAGEMENT OWNERSHIP

Each of the processes described in this response are owned by a Business Excellence Steering Team (BEST) comprised of managers from the appropriate functional area at each nuclear station and the General Office. One of the BEST's purpose is to share information on organizational, programmatic, and team strengths and weaknesses among members to enhance lessons learned. The members are to be cognizant of performance problems which occur and jointly agree on the course of action for resolution of significant issues. The BEST is expected to develop and maintain department level programs and processes related to their functional area. To that end, the BEST is responsible for assuring that the procedures/processes that they own are effective and meet applicable requirements.

6.3 EMPLOYEE QUESTIONING ATTITUDE

Management has encouraged employees at all levels of the nuclear organization to demonstrate a strong questioning attitude. This questioning attitude tends to promote a culture that identifies and resolves low level problems. There have been a number of instances where individual employees have identified problems through the PIP program which resulted in changes in procedures or modifications to SSCs and/or improved understanding of the design basis of SSCs.

6.4 CORRECTIVE ACTION PROGRAM

The corrective action program, as implemented through the various Nuclear System Directives and described in Section 5.0 of this response, is an effective program that is followed at all three of the Duke nuclear stations. The PIP program is the primary tool where non-conforming conditions are captured and tracked. This program captures problems and non-conformances from a variety of sources. The number of items captured in this program has increased over the last several years, which is indicative of increased use of the program and a very low threshold for entering items into the program. Identifying relatively low level problems tends to accomplish two things: (1) resolving minor issues, including process issues, before they become major issues, and (2) allowing trending of minor issues which may identify a need for more extensive corrective action.

The Duke corrective action program has provided the means to provide reasonable assurance that process problems, design issues, operational problems and myriad of other minor issues are identified, evaluated, and corrected. Through the various cause evaluation processes, corrective actions to prevent recurrence are specified and implemented.

6.5 SELF ASSESSMENT PROGRAM

The Self Assessment Program is intended to determine whether activities affecting quality comply with the quality assurance program and whether implementing procedures and processes are effective. These self assessment activities are performed in accordance with approved instructions and procedures (NSD 607, Self Assessments) by organizations independent of the areas being assessed, except for those self-evaluations performed as group assessments. Self-assessment activities are usually both technical and performance oriented, with focus on both the quality of the end product as well as procedures and processes.

Duke Power has two formal programs for quality assurance/technical audits (Regulatory Audits and SITAs) which are discussed in more detail in the following sections.

6.5.1 Regulatory Audits

A system of audits is used to evaluate organizational units conducting quality assurance activities. These audits are performed to determine the effectiveness of implementation of applicable criteria of 10CFR50, Appendix B. Additionally, audits are performed under the cognizance of the Nuclear Safety Review Board (NSRB), Duke's independent safety review committee. The scope of audit activities is reviewed by the NSRB staff prior to audit conduct. The audit staffing is comprised of quality assurance experts and technical subject matter experts having expertise in the activities being audited.

Audits of selected aspects of operational phase activities are performed with a frequency commensurate with nuclear safety significance. The audit process is designed with the intent of assuring that audits of nuclear safety related (QA Condition 1) functions are accomplished within a period of two years. These audits include the assessment of operating, maintenance, and testing procedures.

6.5.2 Self Initiated Technical Audits (SITAs)

Duke's SITA program is a specialized audit conducted to assess the operational readiness and functionality of a QA Condition 1 SSC at a nuclear station using the guidelines detailed in NSD 600, Self Initiated Technical Audits. These audits also give some insight into the effectiveness of the Design Control Program, NSM Program, and corrective action programs when the root causes of findings are examined through the corrective action program. In a SITA, particular attention is directed to the details of plant/system modifications, design control, operations, maintenance, testing, and quality assurance. SITAs include a review of plant procedures in these functional areas. These vertical slice audits are similar in approach and thoroughness to NRC safety system functional team inspections. Appendix B to this response is provided to demonstrate the depth and thoroughness of the effort expended during one of these audits.

These audits are conducted at an approximate frequency of every eighteen months for each nuclear site. Audit scope is established by engineering management from all Duke nuclear sites and support locations. Candidate systems for selection are

generally based on risk significance, system complexity, and experience during the last one to two years, although other considerations may be applied.

The SITA process gives Duke management excellent insight into the design and operation of a number of complex systems at all three nuclear stations. SITAs have identified a variety of issues including:

- UFSAR discrepancies
- DBDs needing to be developed
- incorrect statements in DBDs
- design deficiencies
- field installation problems
- missing documentation and
- testing inadequacies

Since starting this program in 1987, Duke has completed seventeen of these audits as follows:

<u>Oconee Nuclear Station</u>
- *Low Pressure Service Water
- *Electrical Distribution System
- *High Pressure Injection
- *Keowee Hydro (Emergency Power)
- Spent Fuel Pool
- *Emergency Power Distribution System

<u>McGuire Nuclear Station</u>
- *Diesel Generator
- *Electrical Distribution System
- Control Room Ventilation
- *Nuclear Service Water
- *Switchyard

<u>Catawba Nuclear Station</u>
- *Diesel Generator
- 600 VAC Essential Power
- *Electrical Distribution System
- *Auxiliary Feedwater System
- *Nuclear Service Water
- *Component Cooling System

The audits marked with an asterisk (*) indicate that the system audited is considered risk-significant under the guidelines of the Maintenance Rule (for the Electrical Distribution System, only the 4kv portion of the system is considered risk significant). As is evident from the above listing, the majority of the SITAs conducted involved risk-significant system/components.

The SITA process examines the particular system in detail and is intended to identify design basis issues, UFSAR discrepancies, documentation errors, procedural inadequacies and other weaknesses or shortcomings in system design or operation. This critical look at important systems has identified a number of issues which have resulted in plant design changes, improvements in documentation, and improvements in operating practices and procedures. As a result, plant safety has been enhanced and improved understanding of the system's design bases has resulted.

The NRC has recognized the progressive successes within the self assessment process and the SITA process, in particular. Table 6-1 lists excerpts from a number of NRC Inspection Reports from 1989 through 1996 in which NRC inspectors have provided feedback on the Duke Power SITA process. The majority of these comments clearly support Duke Power's belief that the SITA process is an effective tool for performing vertical slice inspections (one of the comments called the results of a particular SITA "mixed").

Problems discovered during a SITA at one site are evaluated for applicability at other sites. The wide range of findings are indicative of the in-depth nature of these audits. Experience has shown that over the years design basis audit findings have evolved from programmatic generic type issues to those of detail specific issues. Duke believes this reflects a healthy environment of continuous fine tuning of design basis documents rather than program development problems found in earlier SITAs.

As an example, the results from the first SITA performed at Oconee in 1987 on the Low Pressure Service Water (LPSW) system concluded that the design basis of the system was not well documented and that there were significant weaknesses in the systems ability to perform its safety function. This was one of the issues that provided the impetus for pursuing the Design Basis Documentation program. Also, as a result of this audit, a number of design changes were made to the LPSW system when the design basis of the system was better defined.

Oconee, being an older vintage plant with several unique design features, has had a number of design review programs for SSCs such as Keowee Hydroelectric Station and the Emergency Electrical Power System. These programs have resulted in a much better

understanding of and availability of design basis information for these systems. Additionally, Oconee has undertaken a review of the safety classification of a number of SSCs which has also contributed to a better understanding of its design basis.

This critical look at SSC design bases through the SITA process provides reasonable assurance that design basis information is understood, that a process exists for uncovering deficiencies in SSC design, and that the stations are operated and maintained in accordance with their design basis.

6.6 SUMMARY

Duke Power believes that the processes and programs, oversight processes, and the self assessment program described in this response are effective and provide reasonable assurance that Oconee, McGuire, and Catawba Nuclear Stations are operated and maintained in accordance with their design bases. When deviations or discrepancies are noted, processes are in place to provide reasonable assurance that the problems are appropriately prioritized, evaluated to determine their extent, and resolved commensurate with their level of significance.

Table 6-1 (Page 1 of 3)
NRC Comments on Duke SITA Process

NRC Inspection Report	NRC Comment
50-413/89-13 and 50-414/89-13	"Strength: A Self-Initiated Technical Audit (SITA) was performed by Duke Power Company Quality Assurance Department on the 600 Volt AC Essential Power System. The audit included an in depth technical review of design documentation, calculations, equipment walkdowns and maintenance procedures. The SITA program is considered a strength."
50-369/89-18 and 50-370/89-18	"It is noted that the Quality Assurance (QA) Department is in a lead role for the Self Initiated Technical Audits (SITA) and previous NRC review has shown these audits to be thorough and valuable."
50-269/89-28, 50-270/89-28, and 50-287/89-28	"The inspectors attended the Oconee Electrical Power System Self Initiated Technical Audit (SITA) exit meeting. The licensee continues to dedicate significant resources and talent in the SITA and Design Basis Documentation (DBD) analysis programs."
50-269/90-24, 50-270/90-24, 50-287/90-24 and 72-4/90-24 (SALP)	"In particular, your Design Basis Documentation Program and Self-Initiated Technical Audit (SITA) were effective in identifying and resolving design deficiencies."
50-369/90-25 and 50-370/90-25	"A self assessment evaluation disclosed a continuing improving trend in the quality of Quality Assurance Department evaluations. A Self Initiated Technical Audit of the Control Room Ventilation System was conducted in 1990. This audit appeared to be very thorough and resulted in 14 Problem Investigation Reports (PIRs - predecessor to the present PIP) being generated."

Table 6-1 (Page 2 of 3)
NRC Comments on Duke SITA Process

NRC Inspection Report	NRC Comment
50-369/93-02 and 50-370/93-02	"The inspector observed portions of the licensee's SITA review of the Nuclear Service Water System. The licensee appeared to be conducting thorough walkdowns; thoroughly evaluating system design and design analysis; and thoroughly evaluating operations, testing and maintenance practices."
50-369/94-13 and 50-370/94-13	"Engineering Self Assessment (40500), a. Self -Initiated Technical Audit - The inspectors interacted with a Self Assessment Team that spent several weeks evaluating two areas of equipment reliability related to engineering, management and maintenance... The inspectors determined that the team conducted the assessment in a systematic, well organized and professional manner. The inspectors concluded that the licensee's audit of the station's switchyard and high voltage station equipment was thorough."
50-369/94-14 and 50-370/94-14	"Self Initiated Technical Audits (SITA) have been initiated to perform in-depth 'vertical slice' audits of systems. A review of SITA on the service water system indicated that the audit was thorough in identifying problems."

Table 6-1 (Page 3 of 3)
NRC Comments on Duke SITA Process

NRC Inspection Report	NRC Comment
50-413/94-17 and 50-414/94-17	"The team reviewed the SITA and compared the team's results to the SITA. The SITA identified numerous electrical and configuration control deficiencies and omissions in instrument and logic testing. Also, the SITA identified inconsistencies between design documents and testing procedures. However, the SITA did not recognize the deficiencies in the design calculations or all the deficiencies in the test procedures (especially the Technical Specification surveillance requirement procedures). Overall the team considered the results from the SITA as mixed."
50-413/95-15 and 50-414/95-15	"Multi-disciplined independent assessments improved. The self initiated technical audit (SITA) process improved through the use of outside expertise between the service water (early in the assessment period) and the component cooling water SITA (end of the assessment period)."
50-269/96-03, 50-270/96-03 and 50-287/96-03	"Duke Power Self-Initiated Technical Audit (SITA) was conducted November 13, through December 12, 1995 of Keowee operational controls, maintenance, surveillance, and other testing to assure Keowee is operated and maintained to perform its safety-related function. The inspector concluded the SITA on Keowee was comprehensive. The inspectors review of the program had essentially the same finding as the SITA."

APPENDIX A

RESPONSE TO REQUEST FOR A DESCRIPTION OF DESIGN REVIEW PROGRAMS

Request: Provide a description of any design review programs including SSCs and plant attributes (seismic, high energy line breaks, etc.). Also, describe how design bases information is maintained current and accessible.

DESIGN BASIS DOCUMENTATION (DBD) PROGRAM INITIATION

Design Basis Documents are a consolidation of the design bases and criteria for a specific plant, structure or system. The Design Basis Documents (DBDs) reflect the current design bases and criteria of that plant, structure or system by including information from, or referencing, other pertinent documents as shown in Figure A-1, "Relationship of DBDs to Other Documents". Design Basis Documents are controlled and kept current pursuant to Duke's Quality Assurance Program.

Duke Power's Design Basis Documentation (DBD) Program was initiated in 1989 as Duke Power identified the need to consolidate and make readily available in an easily retrievable format, design basis information on SSCs. Selected DBDs involved some design basis reconstitution effort as part of the development program. Factors that affected Duke's decision to undertake a design basis documentation program included the results of internal assessments such as SITAs, NRC SSFIs, other regulatory experience, Duke's recognition of the inefficient and time consuming efforts needed while performing operability evaluations and NSM design work, operational experience at Duke Power Company nuclear plants, and industry and regulatory direction.

A pilot program was undertaken to develop a DBD for a selected system at each Duke Power Company nuclear plant. The DBD Pilot Program initiation received the full support of senior department management and was directed by a DBD Steering Committee composed of department management. The DBD Steering Committee objectives included, but were not limited to, the following:

- Definition of scope of the DBD effort,
- Definition of typical content and format of the DBD,
- Definition of SSC and plant design criteria for which DBDs were to be developed,
- Definition of control processes for interfaces, verification, cross disciplinary reviews, sign-offs, and revisions to keep DBDs current,

- Submittal of reports to department management on objective results and lessons learned from the pilot system DBD,
- Publication of the pilot system DBD, and
- Definition of schedule, resources, and cost for the entire DBD effort.

The pilot results were reviewed by department management and the DBD Program effort commenced full scale. Concurrently, plans were made to discuss the DBD Program effort and other design basis issues with the NRC.

DISCUSSIONS WITH THE NRC

The DBD Program was discussed with the NRC on July 28, 1989. The topics covered included:

- DBD concepts
- DBD pilot program
- DBD content, format, scope, project management, and schedule
- Internal initiatives to better manage design issues such as integrated design reviews, design input documentation, document upgrades, test acceptance criteria (TAC), and drawing legibility
- Problem Resolution

An agreement was reached to document findings identified during the DBD Program in accordance with 10CFR 50.73 and NUREG 1022 as supplemented. In accordance with this agreement, seventeen (17) LERs were prepared and submitted during the DBD Program. Problems identified during the DBD development were captured through Duke Power's problem investigation process which includes operability evaluations, reportability evaluations, and necessary corrective actions.

DBD PROGRAM IMPLEMENTATION

Based on the final report submitted by the DBD Oversight Committee, the pilot program completion, and discussions with the NRC, the DBD Program was finalized and implemented. The DBD program generally follows the guidelines established in NUMARC 90-12. Duke Power was represented on the NUMARC Working Group that developed the guidelines. DBD consolidation facilitates review and evaluation of design basis issues by providing information or being a reference source for:

- Design input for NSMs
- Design input for design analyses
- Reference for operability evaluations

- Reference for 10CFR 50.59 evaluations
- Design input to station testing programs
- System function and operation information
- Preparation for or resolution of audits
- Reference for plant life extension
- Cross reference to SSC licensing documents
- Cross reference to regulatory guides
- Training

In general, DBDs were written for:

- each QA Condition 1 system, structure and selected components,
- systems important to normal plant operation,
- systems important enough to capture unique and/or specific licensing commitments, and
- plant specific design criteria such as separation criteria, seismic, fire protection, and single failure criteria.

Standard procedures established the format and content for each DBD. Incorporated in DBDs are test acceptance criteria (TAC) which were developed or consolidated, information regarding SSC operability issues, and unit specific information.

As DBDs were developed and subsequently revised, the design control and design change processes were concurrently revised to include DBDs. DBDs for QA Condition 1 SSCs are controlled as QA Condition 1 specifications and are subject to the Design Control Process as discussed in Section 2.2.2.2. As physical changes to each plant/unit occur, affected DBDs are updated as prescribed by the NSM Program to reflect the changes made and the current design basis and design criteria. Other programs (e.g., Problem Investigation Program, Corrective Action Program) may also initiate a DBD revision. In these situations the DBD is updated as required, either through the NSM Program if appropriate or the Design Control Process.

The DBD Program was completed in December, 1995. Each DBD has a designated owner, e.g., the System Engineer for systems, a Structural Engineer for structures. Tables A-1, A-2, and A-3 list the current Design Basis Documents for Oconee, McGuire and Catawba, respectively. The DBDs for each station are readily available for viewing in electronic form through individual workstations to virtually all plant personnel or in hard copy form from each site's Master File.

Figure A-1 - Relationship of DBDs to Other Documents

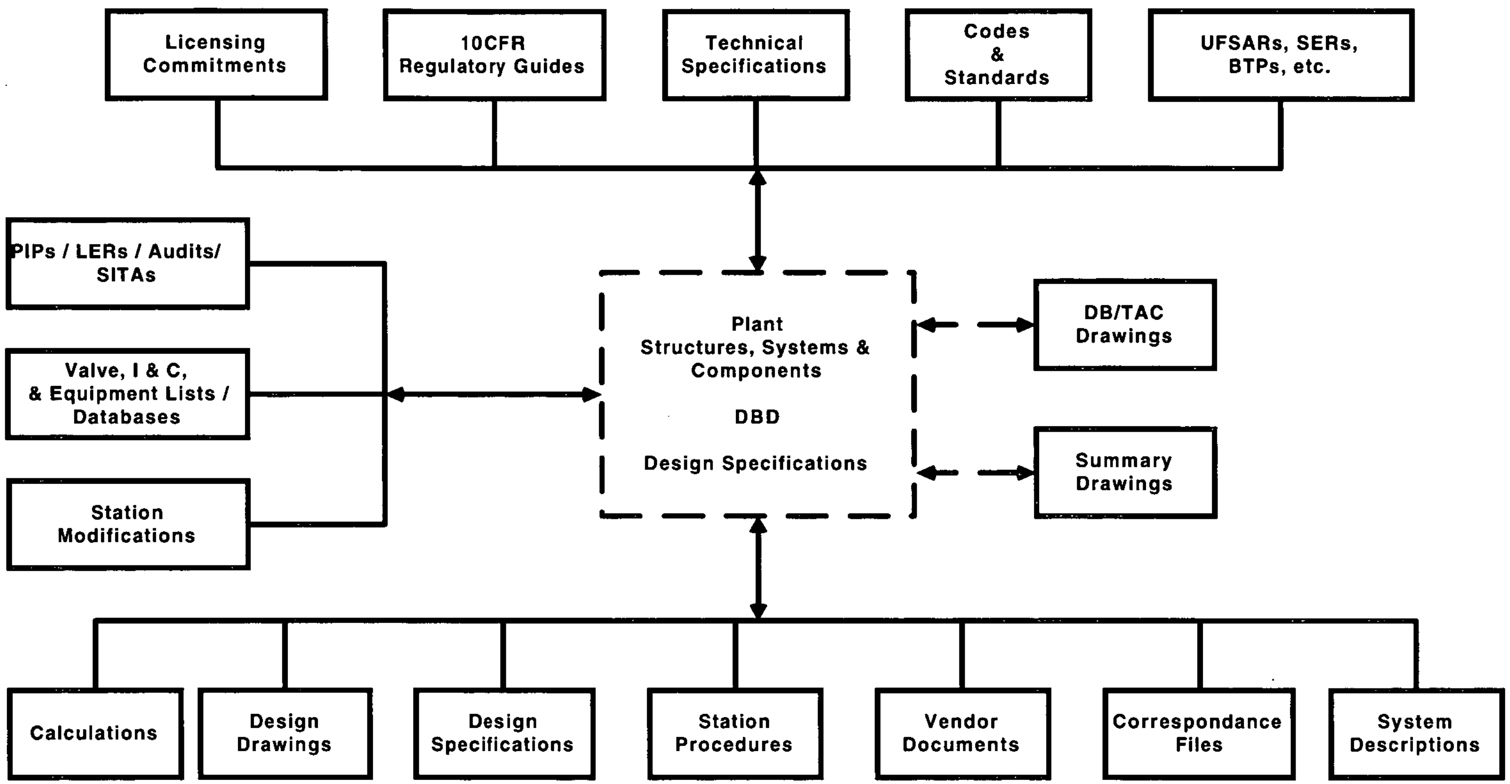

Table A-1: List of Oconee DBDs (Page 1 of 3)

Oconee DBD Title
Auxiliary Shutdown Panel / Loss of Control Room
Cable Tray Supports
Design Basis Events
Fire Protection
Flooding from External Sources
Missile Protection
Oconee Piping Classifications
Oconee Site Systems & Structures, incl. Keowee Hydroelectric
Pipe Rupture / Subcompartment Pressurization
Reactor Building Containment Isolation
Seismic
Selected Plant Design Requirement and Programs: • Electrical Separation • Environmental Qualification • Evacuation Alarm • Meteorological • Setpoints • Pipe Supports • Station Blackout • SSF Structure • ISFSI
Single Failure
230KV Switchyard Structures
Auxiliary Building
CCW Structures
CT4 Transformer and 4KV Switchgear Enclosures
Keowee Structures
Radwaste Building
Reactor Building and Unit Vent
Turbine Building
100KV Alternate Power Supply
120VAC Vital Instrumentation and Control Power
125VDC SSF Auxiliary Power
125 VDC Vital Instrumentation and Control Power
230KV Switchyard
230KV Switchyard 125VDC Power
250VDC Auxiliary Power
4KV Essential Auxiliary Power
4160/600/120V SSF Essential Power
600/208V Safety Related Auxiliary Power

Table A-1: List of Oconee DBDs (Page 2 of 3)

Oconee DBD Title
ATWS Mitigation System Actuation Circuitry & Diverse Scram System
Area Radiation Monitoring
Control Rod Drive
Engineered Safety Feature Actuation
Fire Detection
Inadequate Core Cooling Monitor
Integrated Control
Keowee 125VDC Power
Keowee Emergency Power
Main Control Board
Plant Computer
Process Radiation Monitoring
Reactor Protective System
Security Systems
Auxiliary Building Ventilation
Auxiliary Instrument Air
Auxiliary Steam
Component Cooling
Condensate, Heater Drain, Heater Vent Systems
Condenser Circulating Water
Control Room Ventilation System
Emergency Feedwater and Auxiliary Service Water
Feedwater
Gaseous Waste Disposal
High Pressure Injection and Purification & Deborating Demineralizer
High Pressure Service Water
Hydrogen Purge and Recombiner
Instrument Air
Keowee Air Circuit Breaker (AB)
Keowee Governor Air (AG)
Keowee Governor Oil (OG)
Keowee Turbine Generator Cooling Water (WL)
Keowee Turbine Guide Bearing Oil (GBO)
Keowee Turbine Sump Pump
Liquid Waste Disposal
Low Pressure Injection and Core Flood
Low Pressure Service Water
Main Steam
Penetration Room Ventilation
Radwaste Facility Equipment Cooling
Radwaste Facility HVAC

Table A-1: List of Oconee DBDs (Page 3 of 3)

Oconee DBD Title
Radwaste Facility Liquid Waste
Radwaste Facility Resin Recovery
Radwaste Facility Volume Reduction (Duke)
Radwaste Facility Volume Reduction (Aerojet)
Radwaste Facility Volume Reduction (Stock)
Reactor Building Cooling
Reactor Building Spray
Reactor Building Ventilation
Reactor Coolant
SSF Auxiliary Service Water
SSF Diesel Support
SSF HVAC
SSF Reactor Coolant (RC) Makeup
Service Air
Sluice and Resin Transfer
Spent Fuel Cooling
Spent Fuel Ventilation
Vacuum

Table A-2: List of McGuire DBDs (Page 1 of 3)

McGuire DBD Title
Auxiliary Building Structures
Reactor Building Structures
Nuclear Service Water Structures
Systems Single Failure
System Classification
Containment Isolation
Loss of Instrument Air
Design Bases Events
Radiation Protection
Fire Protection
Seismic
Tornado/Wind
Flood
Loss of Control Room
Electrical Separation
Anticipated Transient W/O Scram Mitigation Actuation Circuitry
Station Blackout Rule
Safe Shutdown System
Offsite Power
EDA - Control Rod Drive Position Indication
EFA - Fire Detection
EHT - Trace Heating
EMB/EMA - Annunciator, Status, and Monitor Lights (EMB)/ESF ByPass Panel (EMA)
EME - Power Monitoring
EMF - Radiation Monitoring
ENC - Source Range Out of Core Instrumentation
EOA/EOB - Main Control Board and HVAC Control Board
EPC - 4.16KV Essential Aux. Power and Protection Relays
EPE - 600 VAC Essential Power
EPG - 120 VAC Vital Instrumentation Control
EPI - 280/120 VAC Essential Aux. Power
EPL - 125 VDC Vital Instrumentation Controls
EPQ - 125 VDC Essential D/G Aux. Power
EQB - D/G Load Sequencer
EQC - D/G Controls
EQD - SSF D/G Instrumentation & Controls
EXA - Plant Security
IEE - Seismic Monitoring
ILE - Pressurizer Pressure & Level Control
ITE - Main Turbine Instrumentation & Controls

Table A-2: List of McGuire DBDs (Page 2 of 3)

McGuire DBD Title
BB - Steam Generator Blowdown
CA - Auxiliary Feedwater
CF - Main Feedwater
FC - Fuel Handling
FD - Diesel Generator Fuel Oil
FW - Refueling Water
KC - Component Cooling Water
KD - Diesel Cooling Water
KF - Spent Fuel Cooling
LD - Diesel Generator Lube Oil
NB - Boron Recycle
NC - Reactor Coolant
ND - Residual Heat Removal
NF - Ice Condenser
NI - Safety Injection
NM - Nuclear Sampling
NS - Containment Spray
NV - Chemical & Volume Control
RF/RY - Fire Protection
RN - Nuclear Service Water
SA/TE - AFWPT Supply/Exhaust
SM/SV/SB - Main Steam Systems
VA - Auxiliary Building Ventilation
VC/YC - Control Room Ventilation / Chilled Water System
VD - Diesel Building Ventilation
VE - Annulus Ventilation
VF - Fuel Pool Ventilation
VG - Diesel Starting Air
VH - Technical Support Center Ventilation
VI - Instrument Air
VN - Diesel Intake and Exhaust
VP - Containment Purge Ventilation
VQ - Containment Air Release & Addition
VS - Station Air
VU/VL - Containment Upper/Lower Compartment Ventilation
VX - Containment Air Return and Hydrogen Skimmer
WC - Conventional Waste Water Treatment
WL/WM - Liquid Waste Recycle
WG - Gaseous Waste Disposal
WN - Diesel Room Sump Pump
WS - Solid Waste Disposal
WZ - Groundwater Drainage
ZD - Diesel Generator Crankcase Vacuum

Table A-3: List of Catawba DBDs (Page 1 of 3)

Catawba DBD Title
Auxiliary Building Structures
Reactor Building Structures
Nuclear Service Water Structures
Systems Single Failure
System Classification
Containment Process Penetrations
Loss of Instrument Air
Design Basis Events
Fire Protection
Post Fire Safe Shutdown
Seismic Design
Tornado/Wind
Flooding from External Sources
Loss of Control Room
Electrical Separation
Anticipated Transient W/O Scram (ATWS)
Station Blackout
Pipe Supports
EB - Switchyard Systems
EFA - Fire Detection System
EHT - Electrical Heat Tracing System
EIA - NSSS Process Instrumentation and Control
EMA - ESF Bypass Panel System and EMB - Annunciator, Status & Monitor Lights System
EME - Reactor Coolant (NC) Pump Power Monitoring System
EMF - Radiation Monitoring System
ENA - In-Core Instrumentation System
ENC - Wide Range Neutron Flux System
EPC - 4.16 KV Essential Auxiliary Power
EPE - 600VAC Essential Auxiliary Power System
EPG - 120VAC Vital I & C Power System
EPL - 125VDC Vital I & C Power System
EPQ - 125VDC Essential Diesel Auxiliary Power System
EPY - 120VAC Essential Auxiliary Power System
EQB - Diesel Load Sequencer
EQC - Class 1E Diesel Controls System
EQD - Standby Shutdown Facility Diesel Generator System
ETL - 600/208/120VAC SSF Auxiliary Power System
ETM - 250/125VDC SSF Auxiliary Power System
EXA - Plant Security-Computer
AD - Standby Shutdown Diesel System
AS - Auxiliary Steam System

Table A-3: List of Catawba DBDs (Page 2 of 3)

Catawba DBD Title
BB - Steam Generator Blowdown System
BW - Steam Generator Wet Lay-Up Recirculation
CA - Auxiliary Feedwater System
CF - Feedwater System
CS - Condensate Storage System
FD - Diesel Generator Fuel Oil System
FW - Refueling Water System
KC - Component Cooling System
KD - Diesel Generator Engine Cooling Water System
KF - Spent Fuel Cooling System
LD - Diesel Generator Engine Lube Oil System
NB - Boron Recycle System
NC - Reactor Coolant System
ND - Residual Heat Removal System
NF - Ice Condenser Refrigeration System
NI - Safety Injection System
NM - Nuclear Sampling System
NS - Containment Spray System
NV - Chemical and Volume Control System
NW - Containment Penetration Valve Injection Water System
RF/RY - Fire Protection Systems
RN - Nuclear Service Water System
SA - Main Steam to Aux. Equipment System and TE - FDWP Turbine Exhaust System
SM/SV/SB - Main Steam Systems
VA - Auxiliary Building Ventilation System
VB - Breathing Air System
VC - Control Area Ventilation System & YC - Control Area Chilled Water System
VD - Diesel Building Ventilation System
VE - Annulus Ventilation System
VF - Fuel Pool Ventilation System
VG - Diesel Generator Engine Starting Air System
VI - Instrument Air System
VN - Diesel Generator Intake and Exhaust System
VP - Containment Purge Ventilation System
VQ - Containment Air Release & Addition System
VS - Station Air System
VV - Containment Ventilation System & YV - Containment Cooling Chilled Water System
VX - Containment Air Return and Hydrogen Skimmer System

Table A-3: List of Catawba DBDs (Page 3 of 3)

Catawba DBD Title
VY - Containment Hydrogen Sample and Purge System
VZ - Nuclear Service Water Pump Structure Ventilation System
WG - Waste Gas Decay System
WL - Liquid Waste system
WN - Diesel Generator Room Sump Pump
WZ - Groundwater Drainage System
ZD - Diesel Generator Crankcase Vacuum System

APPENDIX B

SELF INITIATED TECHNICAL AUDIT (SITA) PROCESS

SITAs are conducted at an approximate frequency of every eighteen months for each nuclear site. Audit scope is established by engineering management from all Duke nuclear sites and support locations. Candidate systems for selection are generally based on risk significance, system complexity, and experience during the last one to two years - although other considerations may be applied.

A team of typically six to ten highly technical, fully experienced subject matter experts is led by a certified lead auditor. Emphasis has recently been placed on increasing the number of non-Duke subject matter experts.

These audits are extensive, with audit planning, preparation, conduct, and reporting generally requiring six to eight weeks to complete. Examples of areas audited include: QA Condition classification decisions, appropriateness of major assumptions regarding system operation and operator actions, adequacy of testing, review and evaluation of operating, maintenance, and test procedures in comparison to the design, system Design Basis Documents, UFSAR compliance, and the appropriateness of design assumptions. System walk downs are conducted during the audit process to assess as-built status relative to design documentation.

Specifically SITAs address:

SSC Design

These audits evaluate the design and documentation against applicable standards. The team reviews the system Design Basis Document as well as applicable sections of the UFSAR, SLC manual, Technical Specifications, and other sources of design basis information. They review applicable industry experience and NRC Inspection and enforcement notices. Past PIPs and LERs associated with the SSC are reviewed to identify recurring problems, verify resolutions are being implemented, and evaluate if the root causes of problems are being corrected.

Operating, Surveillance, and Maintenance Activities

The team reviews applicable startup test results and performance test history. They identify routine operating, surveillance and maintenance activities. These activities are evaluated relative to design requirements. They consider the adequacy of testing to

demonstrate operational readiness. The team does a comparison to vendor information, INPO Good Practices, and test acceptance criteria; examines administrative controls, operability decisions, and procedure history packages; and conducts field observations. Applicable procedures are selected from each group and are reviewed in depth, e.g., operations, maintenance, periodic surveillance, I & E/mechanical preventive maintenance.

Repair/Maintenance Activities

A review of repair/maintenance activities and equipment is completed to evaluate the historical data with regard to maintaining or restoring the SSC to its design configuration, including equipment qualification. Team members also follow through on selected work packages from problem identification to final resolution, including work procedures, compensatory actions, post-maintenance testing, and documentation updates. A review of materials/parts acquisition and commercial grade justifications and documentation is performed.

Nuclear Station Modifications

Several completed NSMs are selected and evaluated from initial design to equipment selection and qualification, vendor documentation, commercial grade justifications, NSM implementing procedures, Project Change Orders, variation notices, as-built field verification, post-modification testing, and follow-up on changes to drawing and procedures. Consideration is also given to whether or not the modification fixed the problem or otherwise satisfied its intended purpose.

Operating Procedures

A review and assessment is conducted to address the adequacy of procedures related to operation during normal, shutdown, and emergency modes. Consideration is given to assumed operator actions from accident analyses and the availability of indications, alarms or other inputs which prompt actions that are significant to safety. The team also considers the appropriateness of assumptions relative to timeliness of operator actions.

Walk-downs

The team performs system walk downs of portions of the system to assess as-built status relative to design documentation and to assess SSC condition and general plant material condition.

Auxiliary Systems

In addition to the activities conducted above, the team also evaluates the contribution of auxiliary systems to operational readiness of the audited SSC.

Qualification and Training

A review is conducted of the qualifications of workers requiring specific authorization or certification to perform work on or affecting the system. The team evaluates whether procedures and practices consider qualifications when assigning work responsibilities. They also assess the adequacy of training programs to provide assurance that defined actions can be carried out.

Source Consideration

The team has at its disposal a large number of documents for review and consideration. These can be generalized as: licensing or regulatory (e.g., UFSAR, QA Topical, Technical Specifications); Programmatic manuals (e.g., NPM and associated directives, Engineering Directives Manual); Procedures (e.g., maintenance, operating); and various modification drawings and vendor drawings, for example.

Audit Report

Audit results are reported to management for review and action. Findings are included in the PIP Program to obtain appropriate review and corrective actions. Findings are reviewed for generic applicability and if appropriate, are assigned to the other Duke nuclear sites for appropriate review and corrective action through the PIP Program.

APPENDIX C

LIST OF REFERENCED DIRECTIVES, WORKPLACE PROCEDURES, AND RESPONSIBLE ORGANIZATIONS

NSD	TITLE	RESPONSIBLE ORGANIZATION
NSD 106	Configuration Management	Engineering
NSD 108	Business Excellence Steering Team (BEST) Process	Senior Vice Presidents and Direct Reports
NSD 202	Event Reporting	Regulatory Compliance
NSD 203	Operability	Engineering
NSD 204	Operating Experience Program	Safety Review
NSD 208	Problem Investigation Process	Safety Review
NSD 209	10 CFR 50.59 Evaluations	Engineering
NSD 210	Corrective Action Program	Safety Assurance
NSD 212	Cause Analysis	Safety Review
NSD 216	Significant Event Investigation Teams (SEIT)	Safety Assurance
NSD 220	UFSAR Revision Process(Under Development)	Regulatory Compliance
NSD 221	Facility Operating License(Under Development)	Safety Assurance
NSD 301	Nuclear Station Modifications	Engineering
NSD 302	Nuclear Procurement Engineering Program (NPEP)	Commodities and Facilities
NSD 308	Plant Operations Review Committee (PORC)	Safety Assurance
NSD 309	Nuclear Safety Review Board (NSRB)	Safety Assurance
NSD 408	Testing	Engineering
NSD 411	Preventive Maintenance Living Program	Engineering
NSD 500	Red Tags/Configuration Control Tags	Operations
NSD 600	Self Initiated Technical Audits	Safety Assurance
NSD 607	Self Assessments	Safety Review
NSD 701	Records Management	Engineering
NSD 702	Document Management	Engineering
NSD 703	Administrative Instructions For Station Procedures	Station
NSD 800	Computer Control	Engineering

APPENDIX C

LIST OF REFERENCED DIRECTIVES, WORKPLACE PROCEDURES, AND RESPONSIBLE ORGANIZATIONS

TITLE	RESPONSIBLE ORGANIZATION
Engineering Directives Manual EDM 101 Engineering Calculations / Analyses EDM 130 Engineering Drawings EDM 170 Design Specifications EDM 210 Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants or the Maintenance Rule EDM 215 Guidance for the Use of Failure Analysis and Trending (FAT) Program to Improve Plant Performance for Systems and Equipment	Engineering
Document Management Manual	Engineering
Nuclear Procurement Engineering Program Manual	Commodities and Facilities
Work Process Manual	Work Control
Compliance Functional Area Manual CFAM 3.3 Facility Operating License/Technical Specifications CFAM 3.7 UFSAR Update CFAM 3.13 Core Operating Limits Report (COLR)	Regulatory Compliance
NGO* Nuclear Engineering Workplace Procedures NE-102 Nuclear Fuel Management NE-103 Documentation of Safety-Related Analyses NE-104 The Safety Evaluation of Fuel Cycle Design and Core Components NE-108 Review/Approval of Externally Supplied Design Documents NE-109 Transmitting Engineering Instructions NE-111 Procurement of Nuclear Fuel, Core Components, and Related Services NE-114 Documentation of Software and Data Quality Assurance Plans XSTP-101 Guidelines for Implementing Fuel/Component Design Changes	NGO* Engineering Manager *Nuclear General Office

APPENDIX D

LIST OF ACRONYMS

ANSI	American National Standards Institute
ATWS	Anticipated Transient Without Scram
BEST	Business Excellence Steering Team
BTP	Branch Technical Position
CCW	Condenser Cooling Water
CFAM	Compliance Functional Area Manual
CFR	Code of Federal Regulations
CMP	Configuration Management Program
COLR	Core operating Limits Report
CT	Current Transformer
DBD	Design Basis Document
D/G	Diesel Generator
EDM	Engineering Directives Manual
ESF	Engineered Safeguard Features
FAT	Failure Analysis and Trending
FOL	Facility operating License
HVAC	Heating, Ventilation, and Air Conditioning
I & C	Instrumentation and Controls
I & E	Instrumentation and Electrical
INPO	Institute of Nuclear Power Operations
ISFSI	Independent Spent Fuel Storage Installation
ISI	In-Service Inspection
IST	In-Service Testing
LER	Licensee Event Report
LPSW	Low Pressure Service Water
LSE	Less Significant Event
MSE	More Significant Event

APPENDIX D

LIST OF ACRONYMS

NE	Nuclear Engineering (in the Nuclear General Office)
NEI	Nuclear Energy Institute
NPEP	Nuclear Procurement Engineering Program
NRC	Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSD	Nuclear System Directive
NSM	Nuclear Station Modification
NSRB	Nuclear Safety Review Board
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
NUREG	NRC Staff Regulatory and Technical Report
OEP	Operating Experience Program
PIP	Problem Investigation Process
PIR	Problem Investigation Report
PORC	Plant Operations Review Committee
QA	Quality Assurance
QC	Quality Control
R & R	Removal and Restoration
SALP	Systematic Assessment of Licensee Performance
SEIT	Significant Event Investigation Team
SER	Safety Evaluation Report
SITA	Self Initiated Technical Audit
SLC	Selected Licensee Commitment
SSC	Structure, System, or Component
SSF	Standby Shutdown Facility
SSFI	Safety System Functional Inspection
TAC	Test Acceptance Criteria
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USQ	Unreviewed Safety Question
VTO	Vital To Operations
WMS	Work Management System