Virginia Electric and Power Company
Richmond, Virginia 23261

February 7, 1997

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

Serial No.	96-535
NL&OS/MAE:	R3
Docket Nos.	50-280/-281
	50-338/-339
License Nos.	DPR-32/-37
	NPF-4/-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
NORTH ANNA POWER STATION UNITS 1 AND 2
RESPONSE TO REQUEST FOR INFORMATION PURSUANT TO 10 CFR 50.54(f) REGARDING ADEQUACY AND AVAILABILITY OF DESIGN BASES INFORMATION

On October 9, 1996, the Nuclear Regulatory Commission (NRC) issued a letter requesting information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information. The letter required submittal of information that will provide the NRC added confidence and assurance that Virginia Electric and Power Company's (Virginia Power) nuclear plants are operated and maintained within their design bases and that any deviations are reconciled in a timely manner. The NRC specifically required submittal of the information described in Items (a) through (e) of the subject letter, as well as information on design reviews or reconstitution programs undertaken for each licensed unit.

Virginia Power recognizes its responsibility to comply with NRC regulations and the facility operating licenses, including requirements regarding design basis information. The design and configuration control processes, design bases review activities, self-assessment program, and corrective action program described in the Enclosure to this letter provide reasonable assurance that North Anna and Surry Power Stations are operated in accordance with their design bases. The information describes current processes and is not intended to reflect future commitments or requirements. Commitments made by this letter are specifically noted below.

The Company has had an extensive Configuration Management and Design Basis Document Program in process since 1989. Preparation of this response has offered an additional opportunity to review the status of these programs and further integrate and focus the Company's efforts regarding design basis information. The scope of the integrated effort includes issuing outstanding design basis documents, completing design basis validation activities, resolution of design basis document open items, and review and validation of the North Anna and Surry Updated Final Safety Analysis Reports (UFSARs).

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The current implementation schedule is to issue the design basis documents (System Design Basis Documents and Plant Design Basis Documents) by June 30, 1999. This schedule commitment supersedes all previous discussions and docketed correspondence regarding the Company's Design Basis Document Program. The Company's plan for UFSAR review and validation will be submitted to the NRC under separate cover.

The Company intends to keep the NRC staff informed of the scope and status of its activities related to the adequacy and availability of design basis information. If you have any questions, please contact us.

Very truly yours,

J. P. O'Hanlon

J. P. O'Hanlon
Senior Vice President - Nuclear

Enclosure

Commitments :

1. The current implementation schedule is to issue the design basis documents by June 30, 1999.
2. Submit a UFSAR review and validation plan to the NRC.

cc: Mr. S. J. Collins
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. P. O'Hanlon, who is Senior Vice President - Nuclear, of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 7th day of February, 1997.

My Commission Expires: March 31, 2000.

Maggie McClure
Notary Public

(SEAL)

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RESPONSE TO NRC RAI RE ADEQUACY AND AVAILABILITY OF DESIGN BASES INFO

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VIRGINIA ELECTRIC AND POWER COMPANY

RESPONSE TO

NRC REQUEST FOR INFORMATION PURSUANT TO 10 CFR 50.54(f)
DESIGN BASES ADEQUACY AND AVAILABILITY

NORTH ANNA POWER STATION - UNITS 1 & 2
SURRY POWER STATION - UNITS 1 & 2

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1.0 Introduction

On October 9, 1996, the Nuclear Regulatory Commission (NRC) Executive Director for Operations issued a letter to the Chief Executives of all NRC licensed nuclear facilities. The letter requested information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information. The request followed several recent NRC findings that indicate industry efforts to improve and maintain design bases information were not effective in all cases.

The NRC requested information that can be used to verify compliance with the terms and conditions of plant operating licenses and NRC regulations, and that the plant Updated Final Safety Analysis Reports properly describe the facilities. This response provides the requested information for the North Anna and Surry Power Stations.

The requested information for items (a) through (e) of the NRC letter is discussed in the main body of this document. The key elements of our engineering design, configuration control, and corrective action processes are common to North Anna and Surry. Station distinctions are made only where appropriate. A summary description, status, and schedule of design review/reconstitution program activities is provided in Attachment A.

2.0 Background

The NRC conducted an Augmented Inspection, a Safety System Functional Inspection, and NRC Resident Inspections at Surry Power Station during the period between September 1, 1988 and March 4, 1989. These inspections resulted in a number of findings that led to an enforcement conference and several violations. Virginia Power outlined corrective actions for the identified inadequacies during the enforcement conference and in response to the associated Notices of Violation. The corrective actions included higher management standards, enhanced sensitivity to events, detailed 10 CFR 50.59 reviews, organizational changes, and initiation of a Configuration Management Program. An operational readiness program, including safety system walkdowns and evaluations, emergency bus and safety equipment power supply testing, a review of work backlogs, and additional functional testing of safety systems, was completed on Surry Unit 1. Key elements of the program were also completed for Surry Unit 2. Those efforts were undertaken to provide increased confidence in the operational capability of the Surry units, including compliance with design bases requirements.

The Company established a Configuration Management Project in 1989 to develop the Configuration Management Program for Surry and North Anna. The purpose of the project was to establish an integrated and systematic management process to control the design, the plant configuration, and the critical station activities that:

- document and maintain the design bases,
- confirm the facility physical configuration is consistent with the design bases, and
- confirm the facility operating documents are consistent with the design bases.

The project was structured to implement that approach through a group of major tasks that included design basis document development, physical verification activities, process analysis, document control improvements, and systems development and integration. The overall goal of these efforts was to provide reasonable assurance that the program objectives would be satisfied in a coordinated manner at both of the Company's nuclear power stations. Further details on the various elements of the Company Configuration Management Project are provided in response to Items (a) through (e) and the attached design review/reconstitution program description.

3.0 <u>Rationale and Effectiveness Considerations</u>

The purpose of the requested information is to provide the NRC with added confidence and assurance that North Anna and Surry Power Stations are operated and maintained within the design bases and that processes exist to identify and reconcile deviations in a timely manner.

The Company's rationale for concluding that plant procedures, configuration, and performance are consistent with the design bases is based, in part, on a reasonable assurance standard consistent with the intent of 10 CFR 50, Appendix A, Criterion 1, and 10 CFR 50, Appendix B. The processes and programs described in the response to Item (a) of the NRC letter require actions intended to achieve that standard.

The Company also implements an ongoing Self-Assessment Program whose goal is to provide the feedback necessary for management to determine, with adequate confidence and assurance, that systems, structures, and components important to safety are operated, configured, and tested consistent with design bases requirements, and that any identified deviations are reconciled in a timely manner. Self-assessment methods relevant to plant procedures, configuration, and performance are discussed in the Company's response to Items (b) and (c) of the NRC letter.

The effectiveness of current processes and programs is evidenced by the results of self-assessment activities and management oversight of performance. These factors, as well as the role of external feedback in determining effectiveness are discussed further in the Company response to Item (e) of the NRC letter. When these processes and programs identified weaknesses, those weaknesses were required to be addressed as warranted.

4.0 Information Request (a)

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

Response

The intent of the Company's Configuration Management policy is to conduct the design, modification, and operation of the North Anna and Surry nuclear stations using engineering, operating, and management principles and controls that address regulatory requirements and industry guidelines. The policy requires integration of interfacing organization activities into an overall program that provides reasonable assurance that plant safety decisions are timely and based on accurate design bases information. The policy has the following primary objectives:

- design and licensing basis maintenance,
- documentation that accurately reflects as-built plant physical and functional characteristics,
- application of design control measures to design changes commensurate with those applied to the original design, and
- control of work activities, including temporary and permanent plant modifications, to address design and licensing basis.

The requirements of 10 CFR 50 Appendix B, 10 CFR 50.59, and 10 CFR 50.71(e) are integrated in procedures that address configuration management related activities.

The information provided here focuses on the common design and configuration control elements that address implementation and maintenance of the approved facility design bases as defined in 10 CFR 50.2. The specific topics discussed are:

- 10 CFR 50.59 implementation,
- design control,
- procedure control,
- Updated Final Safety Analysis Report changes, and
- Technical Specification/Operating License changes.

Other activities that implement the requirements of 10 CFR 50 Appendix B are governed by processes and controls similar to those applied to the above topics. Administrative procedures governing these activities are designed to identify, evaluate, review, and approve changes such that 10 CFR 50.59 safety evaluations are performed when required, the changes conform to design bases requirements, and the licensing basis as described in the North Anna and Surry Updated Final Safety Analysis Reports remains current.

4.1 Implementation of 10 CFR 50.59

The safety evaluation process implements the requirements of 10 CFR 50.59. The administrative procedure governing this process identifies responsibilities and requirements for screening proposed changes, tests, or experiments, performing detailed safety evaluations, and safety evaluation review and approval. The requirements collectively address how to determine if a proposed activity is safe, if it constitutes an unreviewed safety question (or unreviewed environmental question for North Anna), or if the activity requires a change to the Operating License or Technical Specifications. The safety evaluation process also directs the preparer to assess the proposed change against the applicable plant design and licensing basis and provides a method for determining whether a proposed change represents an unreviewed safety question as defined in 10 CFR 50.59. (The corrective action process described below in response to Item (d) of the NRC letter addresses the reportability of deviating conditions requiring safety evaluations).

Activities characterized as changes, tests or experiments are subject to a screening process to determine whether or not a detailed safety evaluation should be performed. If 10 CFR 50.59 is determined not to be applicable to the activity, the activity screening checklist documents that a written safety evaluation is not required. If 10 CFR 50.59 is determined to be applicable to the activity, a written safety evaluation is required. A safety evaluation is required, as a minimum, for the following:

- changes to the Operating License or Technical Specifications,
- temporary modifications (bypasses, jumpers, etc.),
- changes to the facility, procedures, method of operation or alterations to tests or experiments as described in the safety analysis report,
- tests or experiments not described in the safety analysis report,
- deviations that identify discrepancies in the Technical Specifications or in the UFSAR,
- continued operation of non-radioactive systems with detectable levels of contamination outside their design basis as described in the UFSAR (North Anna) or that exceed a limit supported by an approved safety evaluation, and
- changes to the North Anna Environmental Protection Plan.

The completed safety evaluation documents the design and licensing bases considered and the impact of the proposed change on the Operating License and associated conditions, the Technical Specifications and, for North Anna only, the Final Environmental Impact Statement. Procedures require that the form also identify:

- the reason for the safety evaluation and the purpose of the proposed change,
- applicable UFSAR, Technical Specification, safety analysis, design calculation, procedure or other references used in the review,
- limiting conditions or special requirements assumed by the safety evaluation with reference to the formal tracking mechanisms used to satisfy those conditions or requirements,
- the safety implications of the proposed change including a supporting

discussion for each conclusion and the basis for determining whether the proposed change constitutes an unreviewed safety question, and
- safety evaluation reviews, approvals and approval status.

Special qualification and training requirements are invoked by the safety evaluation process. Safety evaluation preparers, reviewers, and approvers are required to meet specific education, experience, and training requirements, including a two week plant systems course, safety evaluation process and preparation training, and annual retraining. (Station Management may exempt individuals from portions of qualification and training requirements on a case-by-case basis).

Safety evaluations are required to undergo the following levels of review and approval:

- independent review,
- Cognizant Supervisor review,
- Design Authority review (for proposed changes involving design basis or design basis programs issues),
- Reactor Engineering review (for proposed changes that potentially affect reactivity),
- Superintendent of Radiological Protection review (for proposed changes that potentially affect airborne activity, occupational exposure, etc.),
- Station Nuclear Safety and Operating Committee review and approval, and
- Management Safety Review Committee review (of a select sample of safety evaluations).

The Station Nuclear Safety and Operating Committee is required to review and approve new procedures and procedure changes that require a safety evaluation, proposed tests and experiments that affect nuclear safety, and proposed changes or modification to plant systems or equipment that affect nuclear safety. The committee is responsible for providing a continuing review of the operational and safety aspects of the stations.

Management Safety Review Committee review is required for changes that are determined to involve an unreviewed safety question as defined in 10 CFR 50.59, a change to the Operating License, or a change to the Technical Specifications. Changes of this nature may not be implemented without prior NRC approval. Management Safety Review Committee activities are discussed further in response to Item (e) of the NRC letter.

4.2 <u>Design Control Program</u>

The Company's Nuclear Design Control Program establishes responsibilities and requirements governing the conduct of design activities. The program addresses the requirements of 10 CFR 50, Appendix B and the guidelines of ANSI N45.2.11-1974 as clarified in The Virginia Electric and Power Company Operational Quality Assurance Program Topical Report (Topical Report). The intent of the program is that:

- applicable regulatory requirements and design bases are correctly translated

into specifications, drawings, procedures, and instructions applicable to design changes and/or modifications,
- changes from specified standards and requirements and the reasons for the changes, are identified, documented, reviewed and approved, and
- appropriate information is documented, reviewed and approved for each design activity.

The Nuclear Design Control Program applies to plant design changes and other engineering activities performed in support of station operation. The program includes controls for developing design inputs, performing design verifications, and preparing design output documents. Various engineering standards are also in place to govern technical program reviews (e.g., Environmental Qualification, Seismic Qualification, Appendix R to 10 CFR 50, Human Factors, etc.) and/or to provide specific instructions on performing certain evaluations or analyses.

Nuclear Engineering Services is considered the Design Authority for the Company's nuclear plants. The Design Authority has overall responsibility for developing and maintaining the plant design bases in accordance with Nuclear Design Control Program requirements. The Nuclear Design Control Program requires design activities to be performed by, or under the direction of, Design Authority personnel including those activities related to design changes.

Two key elements of design control, Design Change and Core Reload Programs, govern activities that design and implement physical changes to the plant including physical changes that can affect the design bases. Descriptions of the Design Change Program and Core Reload Program are provided below.

4.2.1 <u>Design Change Program</u>

A design change package is required to document the design for a plant modification and associated reviews and approvals. The design change package also must include information needed to implement the modification (construction drawings, materials list, etc.), update controlled station documents (including the UFSAR) following implementation, and documentation justifying the change. Procedures require that an individual be designated as the project engineer with responsibility for overall design change package project management from initiation through closeout.

The following specific aspects of the design change package preparation process address design basis considerations and configuration control.

4.2.1.1 Design Change Package Preparation

The Engineering Review and Design section of the design change package is the method used to describe the change, proposed resolution, the design itself, and relevant design bases requirements. Changes to the design bases must be addressed and justified.

Special implementation requirements must be discussed in the Engineering Review and Design section. Their purpose is to provide the implementing organization information that is necessary to properly implement the design change. Examples of special implementation topics include the mode of operation (refueling, cold shutdown, etc.) required for implementation, special provisions and/or activity sequences necessary to address assumptions in the engineering evaluations, and compensatory measures that may be required if barriers (e.g., fire, security, flood, environmental qualification barriers) are breached during implementation. Where interim configurations may exist during implementation or as systems and components are returned to service, the Engineering Review and Design section and the 10 CFR 50.59 safety evaluation are required to address the technical justification and safety implications for those configurations.

Testing requirements must be specified in the Engineering Review and Design section, including the functional and performance testing (and the associated acceptance criteria) required to confirm that the design bases are satisfied. Technical Specification surveillance tests needed to demonstrate operability prior to return to service must also be identified.

Calculations performed in support of a design change are required to be documented, filed in Records Management, and then referenced in the design change package. Each calculation is required to document the design inputs and assumptions commensurate with the requirements of ANSI N45.2.11-1974.

The organization preparing the design change is required to review the Updated Final Safety Analysis Reports, Technical Specifications and, if necessary, prepare the appropriate change requests. The change requests are required to be included in the design change package appendices.

A 10 CFR 50.59 review must be performed for each design change by completing either an activity screening checklist or a written safety evaluation. The applicable document is required to be included in the design change package appendices.

A Controlled Document Summary is required to identify the controlled station drawings, procedures, computer databases, vendor manuals and other controlled documents that require revision for the design change package. Controlled documents affected by the modification must be classified either as priority or non-priority documents. Priority documents are those documents needed to support system or component operation. Priority document updates are required prior to returning systems or components to service following design change implementation and post-installation or pre-operability testing. Non-priority documents are those documents, other than priority documents, required for ongoing station activities and should be updated to reflect the plant as-built configuration. After an initial review by Engineering, groups having expertise in specific areas are responsible for performing reviews and identifying any additional controlled document revisions required.

A Programs Review Checklist is required to be completed and included in the design change package as a record of the programs/topics (Environmental Qualification, 10 CFR 50 Appendix R, Seismic Qualification, etc.) considered in the design. A discussion of program impact must also be included in the Engineering Review and Design section of the design change package. Some programs/topics may require that certain design checklists, document change notices, etc., be completed and included in the appendices of the design change package. The programs review also requires a review of the UFSAR, the Operating License or Technical Specifications, and existing System Design Basis Documents.

4.2.1.2 Review and Approval

The following reviews and approvals are required for design change packages:

- An independent review by an individual(s) in the design organization other than the preparer of the change. The independent review encompasses the entire design change package including the appendices, the Controlled Document Summary, the Programs Review Checklist, and the 10 CFR 50.59 safety evaluation.
- A Project Engineer review for overall adequacy and confirmation that the appropriate engineering disciplines were involved in the independent review.
- A Design Control Engineer review to confirm the design change package conforms to the current approved standards.
- Program owner review and approval, when appropriate, to see that technical program requirements are addressed.
- Station Nuclear Safety and Operating Committee approval of design change packages that involve changes or modifications to plant systems or equipment that affect nuclear safety and/or require a safety evaluation.

Design changes involving unreviewed safety questions or changes to the Operating License or Technical Specifications also require Management Safety Review Committee review. NRC approval is required prior to implementing these design changes.

4.2.1.3 Implementation

Design change packages are normally installed by either Nuclear Site Services or the Maintenance Department. The installing organization is required to use the information provided or referenced in the design change package (including the construction drawings, the materials list, the special implementation requirements, and the appropriate installation specifications) to implement the modification. A Working Field Package is required to consolidate the information and includes the procedures required to implement the modification. The installation documentation is required to be filed with the Work Order(s) or in the Modification Record File for the design change package. Appropriate inspections and supervisory controls provide added assurance the installation conforms with the design.

4.2.1.4 Testing and Return to Service

Either an Operational Readiness Review or a Post Maintenance Testing Test Data Sheet is required for each portion of the modification to be tested and returned to service separately. The purpose of the Operational Readiness Review/Post Maintenance Testing Test Data Sheet is to control, coordinate, and document the following testing and return to service activities:

- verification that the installed modification is consistent with the design,
- identification and proper resolution of open items,
- update and distribution of priority controlled procedures, station drawings, design output documents, vendor manuals and computer databases that require revision prior to commencing functional testing,
- satisfactory completion of functional and performance testing,
- Operations Department acceptance, and
- Station Nuclear Safety and Operating Committee review (if requested).

The Testing organization within Nuclear Engineering Services is responsible for developing a test plan. The test plan must specify the test procedures to be used to address the post-modification testing requirements presented in the design change package. The purpose of post modification testing is to confirm the capability of a modified structure, system, or component to meet specified design parameters. The Testing organization may specify the use of generic test procedures, or existing or new station procedures as appropriate to accomplish the testing objectives.

4.2.1.5 Controlled Document Review and Revision

Following the completion of an Operational Readiness Review or a Post Maintenance Testing Test Data Sheet, a controlled document review and revision form must be initiated to revise the non-priority procedures, drawings, computer databases and other controlled documents that must be updated to reflect the as-built condition. Updates must be completed within a specified time period after initiating the Controlled Document Review and Revision or tracked to manage the responsible organization's commitment to update the document in accordance with applicable program requirements.

4.2.2 Core Reload Program

The Core Reload Program addresses the development and implementation of core reloads. The program is separate from the Design Change Program because of the repetitive nature and limited focus of core reload activities. The Core Reload Program also applies to control rod, burnable poison rod, and fuel assembly design changes.

Core reload design, analysis, testing and certification for operation is addressed by Nuclear Design Control Program implementing procedures controlled and maintained by the Nuclear Analysis and Fuel Department. The reload core design and analysis process is required to be performed in accordance with applicable Topical Report requirements. The program is equivalent in scope and rigor to the design change process discussed above but contains special features that are specific to nuclear fuel.

4.2.2.1 Core Design and Safety Analysis Initialization

The core design and safety analysis initialization process requires a detailed review of the design and licensing environment for the specific station and unit core design. The scope of this review includes:

- the current fuel management scheme,
- applicable correspondence files including information of potential significance to the design basis,
- the applicable Technical Specifications,
- the blank Reload Safety Analysis Checklist that reflects bounding parameters assumed in design basis accident analyses,
- the Restricted Fuel Assembly and Insert Component List,
- active design change packages for the unit being refueled,
- recent generic industry correspondence, and
- active Justifications for Continued Operation and Temporary Modifications applicable to the unit being refueled. These documents may involve the temporary reallocation of performance margins potentially impacting the conclusions of subsequent safety analyses.

The process requires that the review be documented in a Reload Design Initialization Report.

4.2.2.2 Reload Safety Analysis Checklist

The Nuclear Analysis and Fuel Department is responsible for developing and maintaining a list of reload sensitive, safety related inputs used in the currently applicable licensing analysis of FSAR transients. This consists of those accidents whose analyses are presented in the UFSARs. This list is designated as the blank Reload Safety Analysis Checklist and is in the form of a checklist of parameters related to core physics, fuel performance and thermal/hydraulic performance assumed in the current design basis analyses.

For each reload core design, the Nuclear Safety Analysis and Nuclear Core Design groups are required to develop jointly a Reload Safety Analysis Checklist, including cycle specific reactor physics data calculated by the design group, the currently applicable parameter limits, and a notation of cases where the cycle specific parameter exceeds (i.e., is more limiting from a safety analysis perspective) the parameter limit. An evaluation of this information is required to determine the appropriate accidents that must be addressed to support the reload. These evaluations/analyses are required to be documented in calculations and/or technical reports as appropriate, and to be summarized in a Reload Safety Evaluation. Accident analysis models, methods and considerations are governed by the Nuclear Safety Analysis Manual.

Should an accident reanalysis demonstrate that any acceptance criteria derived from the plant design bases are not met, the core may be redesigned, or more restrictive Technical Specifications governing certain operating parameters that cause analysis results to be restored to the acceptable domain may be proposed to the NRC. In some cases these changes may be submitted in the Core Operating Limits Reports. In the event that the criteria are met but changes to parameter analysis limits result, these limits are required to be incorporated into the blank Reload Safety Analysis Checklist for use in evaluating upcoming cycles. In this way, a set of bounding operating core characteristics that has been demonstrated to meet the plant design and analysis basis may be retained from reload to reload.

4.2.2.3 Reload Safety Evaluation

The Reload Safety Evaluation process consists of the following elements:

- completion of the design initialization process,
- completion of the cycle specific Reload Safety Analysis Checklist,
- evaluation of the pertinent thermal evaluation criteria,
- reevaluation/reanalysis of the FSAR transients affected by Reload Safety Analysis Checklist parameters identified as more limiting than the current analysis of record, and
- assurance that the fuel rod design acceptance criteria will not be violated for the cycle.

The Reload Safety Evaluation process requires the following documentation:

- a Restricted Fuel Assembly and Insert Component List for design initialization,
- a Reload Design Initialization Report,
- the cycle specific Reload Safety Analysis Checklist,
- core design and safety analysis calculations that support the safety evaluation,
- a Reload Safety Evaluation Report summarizing the evaluation of core design, safety analysis, fuel performance and Technical Specifications (the core loading plan is also a required element of the report), and
- a Core Operational Readiness Checklist to document the completion of key records that support the manufacture, design, analysis, licensing, loading, testing and operation of the new core.

The Reload Safety Evaluation process also addresses initiation of any required UFSAR changes.

4.2.2.4 Unreviewed Safety Question Determination

Upon completion and documentation of the Reload Safety Evaluation and an interdisciplinary design review, a separate 10 CFR 50.59 safety evaluation is required for the reload design to determine if an unreviewed safety question or a Technical

Specification change is required. Should the results of this evaluation be affirmative, an amendment to the plant operating license is required and the supporting safety evaluations must be transmitted to the NRC for review and approval.

4.2.2.5 Core Operational Readiness Checklist

The Core Operational Readiness Checklist documents the completion of key records that support the manufacture, design, analysis, licensing, loading, testing and operation of the new core. At a minimum, the Core Operational Readiness Checklist must include the following elements:

- Component Design/Fabrication Verification - a summary of significant changes identified during the design, fabrication, offload or onload of the core,
- Safety Analysis Review/Approval - a summary of relevant design, safety analysis and licensing documentation supporting the safety and regulatory requirements for the reload core,
- Core Loading Verification - a summary of any anomalies identified during core onload that could impact the designed operation of the core or associated fuel and components,
- Startup Testing Preparation - a summary record of the preparation and transmittal of technical information designed to support startup physics testing of the core,
- Preparation for Power Operation - a summary record of the preparation and transmittal of technical information designed to support power operation of the core, and
- Follow on Items - summary documentation of the completion of key records required after the return of the unit to power operation (e.g., startup physics test report submittal to the NRC).

4.2.2.6 Development, Review and Verification of Core Loading Plans

The Core Design group is responsible for developing core loading plans for reload cores. The core loading plan requires a detailed core map showing locations of fuel assemblies and insert components by identification number. The core loading plan is required to be based on the final design used in the approved cycle design and safety analysis and be independently reviewed and verified within both the Nuclear Core Design and Fuel Accountability and Inspection groups. Review requirements for core loading plans are established in the engineering manuals for each section.

The Fuel Accountability and Inspection group is required to perform and document an independent verification that the core is loaded in accordance with the approved core loading plan.

4.2.2.7 Reload Startup Physics Testing Predictions and Report Preparation

Operability of the core in accordance with its design basis is confirmed by the Technical

Specifications surveillance program, by the startup physics testing program, and by core performance monitoring. Requirements for the startup testing program address the recommendations of ANSI-ANS/19.6.1, "Reload Startup Physics Test for Pressurized Water Reactors".

Documentation of ongoing core performance monitoring is required throughout the operating cycle via Monthly Core Operating Reports and at the end of the cycle with a Core Performance Report.

The technical basis for the startup physics test and core performance monitoring acceptance criteria is required to be documented in a Reload Design Technical Report.

4.2.2.8 Restricted Fuel Assembly and Insert Component List

The Fuel Accountability and Inspection group is responsible for issuing and maintaining, for each power station, a Restricted Fuel Assembly and Insert Component List to identify fuel assemblies and insert components that have been determined via either manufacturing or operational surveillance to be in a condition that has an unacceptable impact on either incore safety or fuel performance. The list is required to identify, for each fuel assembly and insert component, the reason for restriction or conditional restriction from reuse in cores, any special use limits, and reference documents. The core designer is required to review and be cognizant of the list as part of the design initialization process for each reload core.

4.2.2.9 Nuclear Safety Analysis Manual

Accident re-analyses, including those that support core reloads, are governed by the Nuclear Safety Analysis Manual. The Nuclear Safety Analysis Manual guidance for performing analyses reflects the station design criteria and bases by providing accident-specific guidance to the analyst in the following areas:

- the ANSI/ANS Classification (i.e. Condition II, III or IV) of the event,
- applicable accident analysis acceptance criteria (the criteria reported in the UFSAR as adequate to demonstrate the integrity of the various fission product barriers with the retained margins of safety defined in the Technical Specification bases),
- system initial conditions,
- reactor protection characteristics significant to the specific event and reflect the reactor protection system design basis,
- single failure considerations for the reactor protection/safeguards system that reflect the system design criteria of the UFSAR and 10 CFR 50 Appendix A General Design Criterion 21,
- an overview of key analysis parameters, including core design parameters, to which the accident results are most sensitive and therefore require special attention in ensuring that the accident analysis conservatively envelopes operating conditions, and
- other analysis modeling considerations that may reflect design basis

assumptions about the performance of structures, systems and components important to safety.

Analyses specified in the Nuclear Safety Analysis Manual must be performed with approved computer code input models independently verified to be consistent with the station configuration and design, and treated as controlled documents.

4.3 Procedure Control

The Procedure Control process governs the development, reviews, and approval of new procedures, procedure revisions, procedure changes, and procedure deletions. The Procedure Control process is integrated with the safety evaluation process to apply the requirements of 10 CFR 50.59 to these procedure activities as necessary.

Procedures are typically classified as either administrative or technical procedures. Administrative procedures detail administrative requirements, responsibilities, activities, or actions to implement or address technical programs, and regulatory requirements and commitments. Technical procedures document and detail a series of actions or steps to accomplish a desired objective such as performance testing, maintenance, integrated system operation, or abnormal and emergency response operations.

A multi-tiered review process is required to validate and confirm that new procedures and procedure changes are administratively and technically accurate, consistent with regulatory requirements and commitments, and in accordance with those source documents that define the current design bases. The review process includes the following as required by the nature and scope of the procedure revision or procedure change:

- a writer's guide review to confirm that the procedure format, content, and structure are consistent with the applicable writer's guide, and that the procedure clearly communicates the instructions,
- a technical review to confirm that the procedure addresses applicable regulatory requirements, is technically correct, reflects the as-built plant configuration, reflects proper equipment lineups, provides for system restoration or an evaluation of temporary conditions, and corresponds to applicable design basis requirements,
- a validation review to confirm the procedure is usable, and that the language and level of information is appropriate for whom it is intended,
- a responsible department review to confirm that the procedure methodology and approach is acceptable, and
- an activity screening to determine whether a written 10 CFR 50.59 safety evaluation is required. When written safety evaluations are required, they must be reviewed and approved with the proposed procedure activity.

The validation review may be done using any of the following methods depending on the nature and type of procedure being reviewed:

- Performance Review - actual performance of the procedure (typically on spare equipment),
- Simulator Review - performance of the procedure on the control room simulator,
- Walk-Through - simulating the performance of a procedure without actually affecting equipment,
- Comparison Review - comparing the procedure to a similar approved procedure successfully used in the plant, and
- Table Top Review - reading the procedure and mentally interpreting and evaluating the procedure information to confirm it will be usable when implemented under normal conditions and alternative scenarios.

The administrative procedure governing the Procedure Control process includes guidelines for selecting the appropriate validation method.

Station Nuclear Safety and Operating Committee approval is required for significant changes in accordance with the Technical Specifications.

Approved procedure revisions and procedure changes are required to be submitted to Records Management for controlled distribution or are distributed electronically.

4.3.1 Emergency Operating Procedures

The procedure control process also governs the development, revision and maintenance of emergency operating procedures. The Emergency Operating Procedure Deviation Basis Documents provide the background and supporting information for emergency operating procedures. The document set consists of the Generic Emergency Response Guidelines plus four additional documents specific to each station (North Anna and Surry).

- an Emergency Operating Procedure Setpoint Document discusses the basis for numerical values of process variables for which specific actions are required to be taken in the emergency operating procedures,

- a Step Difference Evaluation Document discusses a step-by-step technical justification of deviations from the generic Westinghouse Emergency Response Guideline for each procedure,

- a Generic Analysis Applicability Document discusses the station specific applicability of the generic analysis performed by Westinghouse to support the development of the Generic Emergency Response Guidelines, and

- a Design Differences Document discusses station specific differences in plant design features from the reference plant used to develop the Emergency Response Guidelines and addresses the implications of the differences.

The Emergency Operating Procedure Deviation Basis Documents comprise the plant-specific technical guidelines defined in NUREG-0899, "Guidelines for the Preparation of

Emergency Operating Procedures". The documents are required to be maintained by the Nuclear Analysis and Fuel Department.

An activity screening is required for changes to emergency operating procedures to determine if the requirements of 10 CFR 50.59 are applicable. If determined applicable, a written safety evaluation is required. Both the affected emergency operating procedure and its background information are required to be evaluated against the 10 CFR 50.59 criteria. Emergency operating procedures also must be evaluated for potential impact on technical programs (e.g., Environmental Qualification, Appendix R, etc.). Lead program personnel are required to be notified to update affected programs as required.

4.4 Updated Final Safety Analysis Report Changes

The Updated Final Safety Analysis Reports (UFSARs) are required to be updated to reflect the currently approved licensing basis for the stations. Changes to the UFSARs are generally the result of revised regulatory requirements, planned changes, or corrective actions. Examples of planned changes include design, Technical Specification, or procedure changes. When a change is required to the UFSAR, the specific change is required to be identified, reviewed, approved, and incorporated in accordance with the UFSAR Change process.

The current UFSAR Change process uses a UFSAR Change Request to document the proposed changes, associated reviews, and approvals. The change request package is required to include a markup of the affected UFSAR page(s), supporting documentation and a written safety evaluation or activity screening checklist. A review of the change request package is required to determine if it :

- is adequately supported by source documentation and the safety evaluation,
- is consistent with NRC commitments and Technical Specifications, and
- satisfies regulatory requirements.

The process requires UFSAR Change Requests to be reviewed by the Supervisor of Licensing, the Design Authority, and the cognizant Department Manager, as appropriate. The Station Nuclear Safety and Operating Committee is required to approve the UFSAR Change Requests.

4.5 Technical Specification Changes

Measures for controlling changes to the North Anna and Surry Technical Specifications are contained in administrative procedures. Technical Specification changes may be requested for a number of reasons, including equipment or system design changes, new fuel or revised accident analyses, and new regulations. Requests for changes to the Technical Specifications and the Facility Operating Licenses must be submitted to the NRC pursuant to 10 CFR 50.90. Technical Specification changes are required to be implemented following NRC approval by the issuance of amendments to the Station's Technical Specifications.

A proposed Technical Specification change must be documented on a change request form. A proposed Technical Specification change request package is required to describe and justify the requested change. The proposed change request is required to be screened and prioritized to confirm that the change is consistent with other Technical Specifications and current NRC requirements. Procedures require that the Technical Specification change request package include a 10 CFR 50.59 safety evaluation, a significant hazards consideration determination, mark-ups of the Technical Specifications showing the proposed changes, revised Technical Specification pages, and any necessary supporting documentation.

The Technical Specification change request package must be reviewed by appropriate departments. The reviews are intended to confirm that the proposed change is in accordance with company policy, addresses the licensing and design bases, and addresses any impacts on plant operations, policies, practices, or procedures. These reviews provide a mechanism to identify actions that will be required to implement the proposed Technical Specification change. These actions are required to be addressed in an implementation plan.

The Station Nuclear Safety and Operating Committee reviews/approves the proposed Technical Specification changes. The Station Manager and the Management Safety Review Committee review/approve the proposed Technical Specification changes prior to NRC submittal.

5.0 <u>Information Request (b)</u>

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

Response

The Virginia Power rationale for concluding design bases requirements are translated into operating, maintenance, and testing procedures is collectively based on cognizance of the plant design bases, and self-assessments of plant procedures and configuration control processes. Together, these elements provide the necessary reasonable assurance as discussed further below.

5.1 Cognizance of the Design Bases

The Company has made several improvements in the adequacy and availability of design bases information in recent years. These improvements have resulted from the Design Basis Document Program initiated in 1989. The program scope, objectives, methods for open item resolution, and status is described in further detail in Attachment A.

The Company's response to Item (a) of the NRC letter explained that the Nuclear Design Control Program is designed to address the requirements of 10 CFR 50, Appendix B and related industry standards as addressed in the Topical Report. The provisions of the Nuclear Design Control Program were established, in part, to provide reasonable assurance that the design bases are correctly translated into plant procedures. Further, the technical reviews required by the procedure control process provide a mechanism for confirming that design bases requirements are addressed as procedures are developed or revised.

5.2 Self-Assessment

The programs and processes described in response to Item (a) of the NRC letter include methods and instructions for translating design bases requirements into plant operating, maintenance, and testing procedures. Company self-assessment activities provide feedback for assessing whether those activities are satisfactorily accomplished.

The self-assessment methods and activities described below have evaluated programs and processes that are required to consider design bases requirements, and have assessed plant procedure conformance to the design bases. Overall, these self-assessments have provided reasonable assurance that plant procedures are consistent with the design bases. Identified weaknesses were required to be documented and corrective actions planned as warranted. Management oversight of plant procedures and related processes is discussed in response to Item (e) of the NRC letter.

5.2.1 Management Self-Assessments

The elements of design and configuration control processes related to plant procedures have been assessed using both compliance and performance-based techniques. Typically, compliance-based assessments review the adequacy of administrative controls and associated processes. This provides added assurance that the design bases are considered, implemented, and maintained in plant procedures as changes occur. The performance-based assessments are designed to confirm that the expected result is achieved. These assessments typically evaluate operating, maintenance, and testing procedures and related procedure changes. Examples of management self-assessments that have evaluated the translation of design bases requirements into plant procedures follow.

5.2.1.1 Technical Specification Surveillance Requirements Reviews

Much of the design bases as defined in 10 CFR 50.2 are embodied in the plant Technical Specifications. Virginia Power conducted systematic reviews of North Anna and Surry Technical Specification surveillance requirements in 1993. The reviews were self-initiated to confirm that Technical Specification surveillance requirements are addressed by station procedures. Categories of surveillances were reviewed including component operability, system lineups, process variable measurements and instrumentation. Identified testing inadequacies were reported to the NRC in accordance with 10 CFR 50.73, and subsequently corrected. The NRC staff monitored this effort and concluded the program was comprehensive, and significantly contributed to improved plant safety.

5.2.1.2 Emergency Operating Procedures Assessment

Assessments of North Anna and Surry emergency operating procedures were conducted in 1990. The assessments included associated abnormal, annunciator response, and operating procedures, the applicable administrative controls, and procedure validation and verification. The overall conclusion was that the Emergency Operating Procedures were consistent with the Westinghouse Accident Response Guidelines and technically sound.

5.2.1.3 Procedure Control Process Assessment

An assessment of the procedure control process was conducted by Corporate Nuclear Safety, Station Quality Assurance, Corporate Procedures, and Operations personnel in 1992. The purpose of the assessment was to determine the effectiveness of the program governing plant procedure upgrades. A sample of technical and administrative procedures were reviewed, including periodic test, preventive maintenance, corrective maintenance, and operating procedures. The overall conclusion of the assessment team was that the upgraded procedures showed improved technical accuracy, level of detail, and human factors considerations.

5.2.2 Safety System Functional Assessments

Select safety systems have been evaluated through safety system functional assessments. The assessments were modeled after the "vertical slice" concept used in NRC Safety System Functional Inspections. While safety system functional assessments are designed to be broad and comprehensive, plant operating, maintenance, and testing procedure reviews are conducted to confirm that design bases requirements are addressed. These assessments have also evaluated plant configuration and performance consistency with the design bases as addressed by Item (c) of the NRC letter. Safety system functional assessments performed by Virginia Power are described below.

5.2.2.1 Electrical Distribution System Functional Assessments

Electrical distribution system functional assessments were performed at North Anna in 1991 and Surry in 1992. The assessments included the electrical distribution system and supporting systems, and the emergency diesel generators and supporting systems. Operating, emergency operating, and testing and surveillance procedures associated with those systems were evaluated to confirm the design bases were effectively addressed. The assessment concluded that overall, the design and operation of the electrical distribution system were acceptable and in the event of a total loss of AC power, the station could be shut down and maintained in a safe shutdown condition until power is restored.

A follow-up assessment was also performed at Surry in 1992. The purpose of the follow-up was to determine the progress of corrective actions identified during the initial assessment. Identified weaknesses from the initial assessments and follow-up were required to be documented and corrective actions planned as required.

5.2.2.2 Service Water System Operational Performance Assessments

Service Water system functional assessments were performed at North Anna and Surry in 1994. The assessments included the service water system and supporting systems. System operating procedures, emergency operating procedures, testing, and surveillance were reviewed. The assessment concluded that overall, the service water systems can perform their design functions and that a sufficient design margin exists to provide added assurance that design requirements are satisfied. Identified weaknesses were required to be documented and corrective actions planned as required.

5.3 Design Bases Validation

Design Basis Integration Reviews provide a mechanism to gain added assurance that design bases requirements are properly translated into operating, maintenance, and testing procedures. Their purpose is to confirm that a system is operated and tested in accordance with its design basis, that the appropriate design requirements (including engineering design requirements) are included in technical procedures, and that the system is capable of performing its design functions. The Design Basis Integration Reviews were previously initiated by the Company but have not been completed as yet. The reviews are currently planned following issuance of system design basis documents as part of the integrated effort discussed in the cover letter of this response. In the interim, the self-assessments discussed above have evaluated the adequacy of existing operating, maintenance, and testing procedures with respect to the design bases.

6.0 Information Request (c)

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

Response

The Company's rationale for concluding that system, structure, and component configuration and performance are consistent with the design basis is based on:

- physical verification, equipment labeling, and systems development activities conducted under the Configuration Management Project,
- verification, testing, and surveillance activities performed during the conduct of operations, and
- self-assessment activities which have addressed system functional and performance characteristics, plant configuration, and design and configuration control process implementation.

These factors collectively provide reasonable assurance of plant configuration and performance adequacy as discussed in further detail below.

6.1 Configuration Management Project

A Configuration Management Project was initiated in 1989 to confirm the baseline plant configuration and strengthen the representation of that configuration in plant documentation. The project included, in part, physical verification and configuration control information systems development tasks. The physical verification task was conducted to provide reasonable assurance that specific controlled station documents accurately reflect the plant configuration. Walkdowns of major plant systems were conducted to confirm that controlled drawings (i.e., flow diagrams and electrical one-line diagrams) correctly reflected plant as-built conditions. The walkdowns also collected component level data for major components. Discrepancies were identified and documented and a corrective action plan established.

The physical verification task also addressed the labeling of plant equipment and plant grid mapping to facilitate equipment identification in documents, and during plant operations activities. The physical verification task was completed in 1994.

The systems development task was conducted to develop or enhance Company information systems to support configuration control activities. The project resulted in the development of the Equipment Data System, Document Management Information System, and Plant Work Control System. Information from physical verification walkdowns was later integrated with the Equipment Data System and additional components were assigned equipment mark numbers in the process.

The processes described in response to item (a) of the NRC letter include the necessary provisions to maintain controlled drawings, equipment identification and labeling, and the data in configuration related computer systems, consistent with plant as-built conditions as design changes occur.

6.2 Operational Activities / Surveillance and Testing

A number of actions required during the conduct of operations and other routine station activities provide added confidence that plant configuration and performance remain consistent with the design bases. Operations shift personnel are required to perform routine general area inspections and equipment checks to monitor system and equipment performance, identify abnormal conditions, and maintain an awareness of system and component operability and configuration status. Other activities such as quality inspections of work activities, post-maintenance testing, in-service inspection program functional checks and pump and valve testing, and Technical Specification surveillance testing provide a means to regularly monitor plant configuration and performance. While these activities are not always directed specifically at design bases requirements, they provide the opportunity for frequent and early detection of system or component degradation which could lead to operation outside the design bases.

6.3 Self-Assessment

The role of self-assessment in providing reasonable assurance was discussed earlier in response to Item (b) of the NRC letter. The self-assessment methods and activities described below have evaluated plant configuration control and performance. Overall, these self-assessments have provided reasonable assurance that plant configuration and performance are consistent with the design bases. Identified weaknesses were required to be documented and corrective actions planned as warranted. Management oversight of configuration and performance issues is addressed in response to Item (e) of the NRC letter.

6.3.1 Configuration Control Assessment

A Configuration Control Assessment was conducted at North Anna and Surry in 1991. The assessment was performance based and addressed removal of equipment from service and return to service, marking and labeling of equipment, and equipment tagging activities. The assessment concluded that existing methodologies were adequate to assure equipment is removed from service consistent with Technical Specification requirements. Recommendations were made to improve independent verifications, administrative procedures for tag-outs, validation of tags in use, and resolution of repetitive tagging deficiencies.

6.3.2 Change Control Assessment

A comprehensive assessment of Virginia Power's change control processes was conducted in 1995. The assessment approach was to:

- review administrative change control procedures to identify potential programmatic problem areas,

- conduct a performance-based assessment by reviewing deviation reports, NRC inspection reports, audits, assessment reports, and isolated problems attributable to change control, and
- interview personnel to identify and/or confirm programmatic change control issues.

The assessment was a broad spectrum analysis covering activities that could alter or make different some aspect of the design, function, operation, or physical configuration of the power stations.

The assessment team concluded that, in general, the existing processes for controlling changes were adequate and have been effectively implemented. No safety significant issues were identified.

6.3.3 Safety System Functional Assessments

The configuration, function, and performance of select safety systems have also been assessed through safety system functional assessments. Assessments performed by Virginia Power were presented earlier in response to Item (b) of the NRC letter.

7.0 Information 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.

Response

Virginia Power nuclear management has emphasized the importance of a questioning attitude to encourage employee or contractor identification of problems discovered in daily station operations. Routine activities such as work inspections, operator rounds, use of procedures, functional and surveillance testing, industry experience reviews, and self-assessment provide the opportunity for problem identification.

The corrective action process consists of four major elements:

- deficiency identification,
- deficiency evaluation,
- corrective action implementation, and
- effectiveness verification.

A problem or deficiency may be documented initially by deviation reports, Nuclear Oversight findings, the Minor Maintenance Program, work requests, or the Industry Operating Experience Review Program. The Potential Problem Reporting System is required within Engineering to evaluate complex design concerns not yet determined to be deviating conditions. Open items identified during design basis document development are required to be addressed in accordance with this process as discussed in Attachment A.

Corrective action effectiveness verification is similar for the various methods of problem identification. The method for reporting a problem or deficiency to the NRC is addressed in deviation report evaluations.

7.1 Deviation Reports

The deviation report is used to identify potentially adverse conditions to the Operations Department Shift Supervisor and station management. The deviation report is also the mechanism for determining external reporting requirements for immediate reportability and reportability within 30 days (10 CFR 50.73) as required by regulations and the Technical Specifications.

Deviation reports may be initiated by any employee or contractor who discovers an unexpected or abnormal condition. The threshold for submitting deviation reports is kept at a low level to encourage problem identification. Specific examples of conditions requiring a deviation report are provided in the administrative procedure that controls this process. Several examples relate to design bases issues.

The procedure for submitting deviation reports requires that the deviation report be immediately brought to the Operations Shift Supervisor, who is a Senior Reactor Operator (SRO). This provides for a prompt operability and reportability review. The Shift Supervisor is required to determine adverse effects on systems, structures, and components and may enlist support from the Shift Technical Advisor, Nuclear Engineering Services, and other station personnel as necessary. Governing procedures incorporate the guidance provided by Generic Letter 91-18, "Information To Licensees Regarding Two NRC Inspection Manual Sections On Resolution Of Degraded And Nonconforming Conditions And On Operability".

The procedures also require categorization of each significant condition as a reportable or non-reportable deviation. A reportable deviation is a deviation that must be reported to the NRC. The reporting requirements differ for each of the categories of deviation but procedures require notifying the appropriate levels of management in each case.

Deviation report deficiency evaluations are required to be performed in accordance with the Root Cause Program. This evaluation includes determining the significance level, the frequency of occurrence using past experience, deviation report trends and a search of one or more station databases (e.g., Plant Work Control System). The appropriate level of root cause evaluation is then determined.

A Category 1 root cause evaluation is required for significant deviations and potentially significant deviations that occur frequently. This consists of a detailed systematic approach to determine the root cause(s) and contributing factors for human/programmatic or equipment performance problems. The results are provided to management in a formal report.

A Category 2 root cause evaluation is required for potentially significant deviations which occur occasionally, as well as routine deviations that occur frequently. This consists of a systematic approach to determining the apparent cause(s) and recommended corrective action of human/programmatic or equipment performance problems. The results are provided to management in written form.

A Category 3 root cause evaluation is required for routine deviations which occur occasionally or infrequently. This consists of determining the cause of human/programmatic or equipment performance problems by reviewing immediately available material or information. The results are documented in the response to the deviation report.

Corrective actions are required to be implemented and documented in accordance with approved management processes. The assigned department is required to develop a corrective action plan. At least two individuals in the Station Nuclear Safety Department are required to review the corrective action plan. One of the individuals must have Reactor Operator, Senior Reactor Operator, or Shift Technical Advisor training at the respective station. Corrective action plans must be reviewed and approved by the Station Nuclear Safety Department or the Station Nuclear Safety and Operating Committee depending on significance. Significant corrective action completion is required to be tracked by the station commitment tracking system. Other corrective action closures are required to be tracked internally by the responsible department.

7.2 Potential Problem Reporting System

Engineering maintains a program to evaluate complex design concerns identified by any individual within the Engineering organization that, once evaluated, may be determined potentially adverse. The Potential Problem Reporting System allows for detailed, multi-discipline reviews of identified concerns and provides a mechanism to document the management resolution. As with other problem identification methods, procedures require Potential Problem Reports determined to be deviating conditions to be documented in a deviation report.

7.3 Nuclear Oversight Findings

The Internal Audit Program addresses, through investigation, the adequacy of and adherence to established procedures, instructions, specifications, codes, and standards, or other applicable contractual and licensing requirements, and the effectiveness of implementation. Nuclear Oversight findings address deficiencies and non-conformances documented during Quality Program Oversight activities. When findings are identified, the deficiency must be investigated to determine the cause, extent, and corrective actions necessary to resolve the deficiency. Deficiencies that have an impact on equipment, operations, or raise a potential operability or reportability concern are required to be documented in a deviation report.

The effectiveness of corrective action in preventing recurrence is required to be monitored. Nuclear Oversight is required to perform follow-ups by obtaining responses and evaluating adequacy, assessing corrective action progress, and that the corrective action meets the intent, and closes findings as appropriate. If it is determined that the response to an internal audit finding is unacceptable, if a finding response is not received in the time allotted, or if corrective action for a finding is not accomplished as indicated on the response, the matter is required to be escalated to the Station Manager or appropriate Corporate Manager for resolution. Appropriate levels of management must be notified if Nuclear Oversight does not agree with the resolution proposed. Findings must be escalated to either the cognizant vice-president or the Senior Vice President-Nuclear when:

- the finding response is late or unacceptable,
- inadequate corrective action is past the due date,
- corrective action taken as of the due date is incomplete or inadequate, or
- a previously identified finding recurs.

Audits are closed following satisfactory closure of associated findings.

The Manager - Nuclear Oversight is assigned responsibility for analyzing audit reports for trends and corrective action effectiveness. As trends are discovered or if the effectiveness of a program is in question, the analysis of the Manager - Nuclear

Oversight is required to be forwarded to the management level consistent with the seriousness of the problem.

7.4 Work Requests

Work requests are required to document hardware deficiencies noted during routine activities such as periodic testing, inspections, operations, maintenance, and observations. A work request is the mechanism required to initiate corrective action for maintenance on permanently installed equipment, work required on items in storage, and repairs to security-related equipment. As stated above, potentially adverse conditions require a deviation report. This assures such conditions can be evaluated and appropriate corrective actions addressed.

7.5 Minor Maintenance Program

The Minor Maintenance Program addresses deficiencies such as tightening of flanges to stop leakage, or similar maintenance items. This program is applicable to maintenance that does not normally require equipment tag-outs or detailed procedures. Minor maintenance works along with other station processes to maintain plant systems, structures, and components in their as-designed condition.

7.6 Industry Operating Experience Program

Industry operating experience evaluations are required to determine potential applicability to Virginia Power. The industry information reviewed includes INPO Significant Operating Experience Reports, INPO Significant Event Reports, NRC Generic Letters, NRC Bulletins, NRC Information Notices, Westinghouse INFOGRAMS, Westinghouse Nuclear Safety Advisory Letters, and vendor 10 CFR 21 notifications. If the review indicates that corrective actions may be required, a report is required to be prepared to fully address the industry issue. Corrective actions developed as part of the review are required to be assigned to a responsible department, tracked and verified. During this process any issues that are determined to pose operability or reportability concerns are required to be documented on a deviation report. This permits prompt review by the plant staff. A formal operating experience review is also required for Licensee Event Reports and Notices of Violation issued from either station to determine if a similar condition exists at the other station.

7.7 Effectiveness Verification

Corrective action verification may be accomplished by station self-assessment, the Nuclear Oversight Internal Audit Program, deviation report tracking and trending, close-out of Commitment Tracking System items, and other management activities such as the Integrated Trending Program. The results are required to be reviewed by management and appropriate corrective actions identified and implemented.

8.0 Information 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.

Response

Virginia Power employs a multi-tiered self-assessment approach to determine the effectiveness of current processes and programs in achieving plant configuration and design bases consistency. Self-assessment methods and relevant examples were presented in the Company's response to Items (b) and (c) of the NRC letter. The goal is to provide the feedback necessary for Company management to determine, with reasonable assurance, that systems, structures, and components important to safety are operated and tested consistent with design bases requirements and that any identified deviations are reconciled in a timely manner.

Management addresses the overall effectiveness of current processes and programs by monitoring self-assessment results and performance trends. The techniques used to evaluate and monitor performance are discussed below. Collectively, these performance indicators have provided reasonable assurance that current processes and programs are effective.

8.1 Management Oversight

The management self-assessment process is designed to maximize station operational safety and optimize the effectiveness of individuals, departments and stations by systematically comparing performance to established standards. Actions are taken to improve performance where warranted. The process has program elements for monitoring individual, departmental and integrated station performance.

Managers are responsible for developing and implementing assessments for activities and programs under their cognizance. The integrated station level assessment process includes quarterly evaluation of the station annunciator windows, deviation report trending, periodic team assessments, and Category 1 Root Cause Evaluations assigned by station management.

The Station Annunciator Windows Program is a tool for management to monitor performance issues and concerns by providing an evaluation of performance against established criteria. Performance reports graphically present current performance status using color coded panels for each performance category similar to control room annunciator windows. The panels are green if the assessed area is a significant strength, white if satisfactory, yellow if needing improvement, or red if a significant weakness. For the latter two categories, corrective actions are required to be developed and reviewed by management. While the windows integrate information to provide a broad perspective of effectiveness, management uses individual indicators such as the relative number of Licensee Event Reports, non-compliances, weaknesses, and human error data that are identified by internal and external sources. These provide specific data related to deviations from Technical Specifications, design bases, and procedure adherence to requirements.

Deviation report trending includes trend categories for change management, design basis issues, and equipment and component performance. Overall, these performance indicators have shown that current processes and programs are effective.

8.2 Nuclear Oversight Audits

The system of audits was devised to assess quality related aspects of the power stations. Internal audits are performed on a frequency commensurate with safety significance and in a manner that assures that audits of safety related activities are completed at least once every two years. Additional audits may be performed as deemed necessary by management to determine the overall effectiveness of processes and programs related to the design and licensing bases.

Provisions have been established that require audits to be performed in those areas where the requirements of 10 CFR 50, Appendix B should be implemented. This includes activities associated with station operation, maintenance, modification, design changes, and operating and test procedure implementation.

Management is required to respond to each audit and initiates corrective action where indicated. Nuclear Oversight conducts documented follow-ups. Where ineffective implementation or a declining trend warrants additional management attention, that fact is required to be noted in the audit report for further action.

The audits listed below are examples of those performed in 1995 and 1996 relevant to plant configuration control activities and programs. In general, these audits have concluded that current programs are effectively implemented. Findings identified by these audits were addressed in accordance with the corrective action process.

8.2.1 Inservice Inspection Audit

The Inservice Inspection Audit conducted in 1995 evaluated the implementation of the applicable requirements of the Technical Specifications, 10 CFR 50 Appendix B, and the ASME Boiler and Pressure Code, Section XI. The Component Examination, Component Support, Pump Inservice Testing, Valve Inservice Testing, Repair and Replacement, System Pressure Test, and Steam Generator Tube Examination Programs were included in the audit scope.

The audit concluded overall that the Inservice Inspection Programs meet and effectively implement regulatory requirements. Performance improvements were noted in the areas of code interpretations, design control, and transmittal of nondestructive examination records.

8.2.2 Technical Specifications and License Requirements Audit

Selected safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, applicable license conditions, and administrative requirements were assessed for translation into procedures and implementation. As much as possible, observations of ongoing activities were conducted to evaluate personnel performance.

Based on the audit activities, it was concluded that, overall, the programs for meeting Technical Specifications and License Conditions are effectively implemented at both stations in accordance with regulatory commitments and industry guidelines. From a performance perspective, it was concluded that Engineering, Operations, Maintenance, and other support organizations work well with the established programs to assure that the station is operated within the limits of the Units' Licenses, Technical Specifications and the Updated Final Safety Analysis Report.

8.2.3 Operations/Refueling Activities Audit

This audit evaluated the effectiveness of Operations Department activities at North Anna and Surry with a focus on safe operation of the reactor, and protection of the core during refueling. Portions of the audit evaluated configuration control related activities such as test control, operating status, conduct of operations, and temporary modifications.

The audit results indicated that, overall, the Quality Assurance Program requirements are effectively implemented at Surry and North Anna.

8.2.4 Design Control and Engineering Programs Audit

The audit evaluated 10 CFR 50, Appendix B Quality Assurance criteria and other program attributes such as document control, test control, corrective actions, design change packages, field changes, engineering transmittals, and installation problem reports. Safety evaluations associated with sampled design change packages were also evaluated.

The audit results indicated that, overall, the Quality Assurance Program requirements related to the Design Control and Engineering Programs are effectively implemented.

8.3 Management Safety Review Committee

The Management Safety Review Committee provides an offsite safety review function and an independent review of designated activities delineated in the station Technical Specifications. The Management Safety Review Committee activities that provide an indication of process and program effectiveness include review of:

- a sample of 10 CFR 50.59 safety evaluations,
- proposed changes to the Technical Specifications or Operating Licenses,

- violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance,
- significant operating abnormalities or deviations from normal and expected performance of unit equipment that affects nuclear safety,
- events requiring written notification to the NRC,
- recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety, and
- a representative sample of reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

The Management Safety Review Committee also provides an independent review in the areas of station operations, maintenance, reactivity management, engineering, chemistry and radiochemistry, radiological safety, quality assurance practices, and emergency preparedness.

8.4 UFSAR Reviews

Virginia Power has previously conducted programmatic reviews of the North Anna and Surry Updated Final Safety Analysis Reports (UFSARs). Discrepancies identified during the reviews were addressed by the preparing and processing individual UFSAR change packages as appropriate. Whenever appropriate, deviation reports were required for identified discrepancies. The deviating conditions were evaluated for operability and reportability, and corrective actions were initiated in accordance with the corrective action process.

Most recently, a UFSAR Project Team was formed in May 1996. The team has examined the existing administrative controls for maintaining the UFSARs, as well as UFSAR content and usability. The assessment results did not identify any safety significant findings. However, the need was identified for several process improvements directed at simplifying administrative controls, increasing owner accountability for technical content, and improving UFSAR access and usability through conversion to an electronic format.

The Company is also assessing the overall accuracy of the North Anna and Surry UFSARs. Assessments are being conducted by teams with operations, design & system engineering, and licensing expertise on an integrated basis in accordance with the methodology outlined in NEI 96-05, "Guidelines for Assessing Programs For Maintaining the Licensing Basis". Although the results to date have not indicated significant safety concerns, errors and inconsistencies have been identified. The team is required to address any identified weaknesses during the course of the review. The overall assessment results will be evaluated and appropriate actions taken through the ongoing UFSAR improvement initiative. Future plans for UFSAR review and validation actions will be submitted to the NRC under separate cover.

8.5 Safety Evaluation Assessments

Safety evaluation assessments were performed by the Corporate Nuclear Safety group in 1992, 1993, and 1995.

The scope of the 1992 assessment included an evaluation of:

- training and qualifications for safety evaluation preparers and activity screeners,
- the definition of organizational and individual responsibilities and interfaces required for a thorough safety evaluation process,
- the adequacy of the procedural guidance itself, and
- the adequacy and quality of evaluations performed to date using the process.

The assessment team concluded that the program met the intent of 10 CFR 50.59 and was an effective tool for reviewing changes to determine if an unreviewed safety question exists. The team further concluded that when individual procedures that control changes invoke the safety evaluation procedure, the safety evaluation and activity screening program was being implemented adequately at both Surry and North Anna and by Corporate groups responsible for performing safety evaluations. There were no unreviewed safety questions or safety issues identified by the assessment team. A follow-up assessment performed in 1993 produced similar conclusions.

A second follow-up Safety Evaluation Program Assessment was performed in 1995. The assessment team:

- reviewed the results of safety evaluation independent reviews to confirm their effectiveness in identifying unreviewed safety questions and nuclear safety issues,
- evaluated the use of the activity screening checklist to confirm that safety evaluations are being performed for those activities required by 10 CFR 50.59 and by Virginia Power administrative processes, and
- confirmed that issues raised in previous assessments were effectively resolved.

The 1995 assessment concluded that the Safety Evaluation Program is effective and that the process provides adequate assurance that facility changes are evaluated for unreviewed safety questions or significant safety issues. Recommendations from previous assessments were determined to be effectively implemented.

8.6 External Feedback

Evaluations and inspections of plant configuration and performance are periodically conducted by external organizations such as the NRC, INPO, and others. These activities provide an indication of internal self-assessment program effectiveness but are not relied on to identify and correct problems or further develop good practices. As evidenced by the most recent SALP Reports for North Anna and Surry, current processes and programs have been effective overall. Weaknesses identified by external organizations are required to be tracked and corrected in a timely manner.

8.7 Overall Conclusion

In summary, the results of the Virginia Power multi-tiered self-assessment program as described above have provided reasonable assurance that the North Anna and Surry configurations are consistent with their design bases.

9.0 Design Review/Reconstitution Programs

In responding to items (a) through (e), indicate whether you have undertaken any design review or reconstitution programs, and if not, a rationale for not implementing such a program. If design review programs have been completed or are being conducted, provide a description of the review programs including identification of the systems, structures, and component(s), and plant-level design attributes (e.g., seismic, high-energy line break, moderate-energy line break). The description should include how the program ensures the correctness and accessibility of the design bases information for your plant and that the design bases remain current. If the program is being conducted but has not been completed, provide an implementation schedule for structures, systems, and components and plant-level design attribute reviews, the expected completion date, and method of SSC prioritization used for the review.

Response

Virginia Power has undertaken a significant effort in recent years to address the correctness and accessibility of the plant design bases. The centerpiece of this ongoing effort is the Design Basis Document Program. The documents issued and under development address both system design bases as well as plant level design attributes. This program is described further in Attachment A.

The Design Basis Document Program is not yet complete. Work remains to finalize the full complement of system and plant design basis documents, and the balance of the integrated effort to complete design basis validation efforts, resolve open items and convert the documents to an electronic format. The electronic conversion is intended to improve design basis document usability and broaden accessibility to the information contained in the design basis documents. The Company's commitment for the system and plant design basis document portion of the integrated effort is addressed in the cover letter for this response. The current implementation schedule for issuing the design basis documents is presented in Attachment A.

ATTACHMENT A

Summary Description

Of

Virginia Power's

Design Basis Document Program

And

Related Engineering Support Activities

Table Of Contents

1.0 **Introduction**

This document summarizes and describes Virginia Power's Design Basis Document (DBD) Program. The DBD Program is part of Virginia Power's overall Configuration Management Program initiated in 1989 as a formal effort to:

- review, document, and reconstitute, as necessary, the North Anna and Surry Power Stations' design bases,
- enhance related engineering support programs, and
- integrate the design bases into a comprehensive configuration control process to ensure that appropriate plant documents reflect the design bases and as-built conditions.

The initiative for the Configuration Management Program is predicated on Nuclear Regulatory Commission (NRC) and industry findings during the late 1980s that many utilities did not have a complete understanding of their nuclear plant design bases and that the necessary configuration control was not always maintained.

1.1 Scope and Objective

The objective of the DBD Program is to identify, assimilate, consolidate, and document the design bases of the Company's nuclear power stations, Surry and North Anna. The DBD Program, in conjunction with other configuration management activities, establishes the design bases for the plants and a means for maintaining and controlling changes to the plant's design bases. The scope of the project includes the following major tasks:

- Completion of a Design Basis Document Assessment Program (DAP) to identify, collect, organize, and index documents that contain significant amounts of design basis information. The product of this program is the Design Basis Document Library Series (DBDLS).
- Preparation of System Design Basis Documents (SDBDs) for the major systems that comprise a nuclear plant (approximately 50 systems per plant).
- Preparation of a Plant Design Basis Document (PDBD) for each nuclear power station that defines the design bases of features and structures that are common to the units or multiple systems.

2.0 **Design Basis Concepts**

The DBD Program is a comprehensive effort to document the engineering design bases for North Anna and Surry. There are a number of concepts that are applicable to the program that often form the basis for frequently asked questions about the program. Some of the major concepts incorporated into Virginia Power's DBD Program and their implementation are briefly discussed below.

2.1 Design Bases Definition

The legal definition of "design bases" is contained in 10 CFR 50.2. This definition is not sufficiently broad or comprehensive to be the basis of a comprehensive design basis document based on the Company's approach to the design basis issues that have been identified. The Company's definition is more consistent with the definition of "engineering design bases" presented in NUREG-1397. The Company's approach is consistent with a "comprehensive" design bases document as outlined in NUMARC 90-12. When the term "design bases" is used in the context of a DBD in this document, it should be interpreted to mean the more comprehensive term "engineering design bases."

2.2 Licensing Basis

The "licensing basis" is a significant part of the engineering design bases of a nuclear plant. As used in the Company's program, the licensing basis includes any docketed documents and information that identify a commitment or obligation to perform certain actions relative to the design of the plants. The licensing basis is documented in a number of docketed documents, including, but not limited to, the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, Operating License, NRC Safety Evaluations, response to bulletins, generic letters, etc., and other correspondence. The identification of the licensing basis of a plant is a fundamental requirement for developing DBDs.

The Company's SDBDs identify and address specific licensing basis requirements for a system in Chapters 8.0 and 13.0 of each SDBD.

2.3 Design and Engineering Documents

Design and engineering documents are considered to be that existing collection of design documents as prepared by the design authority in support of the design functions identified in Appendix B to 10 CFR 50 and ANSI N45.2.11. These documents consist of "input," "process," and "output" documents as identified in NUMARC 90-12 and ANSI N45.2.11, and are the official documents that have historically identified the engineering bases of the nuclear plants. Prior to the initiative to formally document the design bases of the plants, the design and engineering documents served to establish and document the design bases of the plants. The initiative to develop DBDs formalizes the presentation of the engineering design bases in a consistent format, however, the "design and engineering documents" maintained by the design authority constitute the Appendix B to 10 CFR 50 and ANSI N45.2.9 design records. The development of a DBD is the compilation, assimilation, and organization of this "design and engineering information" into a directory-type document.

3.0 Document Assessment Program

An integral part of the DBD Program was the Document Assessment Program (DAP).

Since DBDs are a compilation and assimilation of existing design and engineering information, DBD development relies on the availability of reference sources that contain design basis information and supporting engineering information. The goal of the DAP was to identify, collect, and organize design and engineering information so that it would be readily available to DBD developers in a computerized application.

The initial effort to locate, collect, organize, scan, enter data, and index documents was included in the DAP. The DAP was a one time effort that evolved into the Design Basis Document Library Series (DBDLS). The DBDLS is a computer (PC) based application that uses CD-ROM technology to store data and images. Documents containing design and engineering information are included in the DBDLS. These documents are organized and indexed so that they can be easily researched for DBD development. The DBDLS content has been updated since initial development to incorporate additional design and engineering information documents.

4.0 Licensing Basis

4.1 Introduction

The "licensing basis" for a plant is an integral part of both the "design bases" of the plant as defined in 10 CFR 50.2 and the "engineering design bases" as established in the Company's DBDs.

The identification of the current licensing basis of a plant is a fundamental part of DBD development. At the outset of the Company's initiation of the DBD Program in 1989, a logical systematic approach was established to identify the current licensing basis of the plants and to make it available to DBD developers. As a prerequisite to this task, it was necessary to define what is meant by the licensing basis of the plant. The licensing basis was conservatively considered as any written correspondence between the licensee and the NRC as contained on the docket in the NRC's Public Document Room. A "license commitment" was defined as a statement that contains requirements to be performed, or actions to be implemented, during the design, construction, testing, operation, maintenance, or other activity conducted pursuant to the NRC's operating license as contained on the docket.

4.2 Licensing Commitment

A "Licensing Commitment Assessment Task" was initiated as part of the Document Assessment Program discussed above. The objective of the "Licensing Commitment Assessment Task" was to identify the licensing commitments that were made to the NRC and that partially constitute the licensing basis for the plant. The scope of this task was to identify and review selected licensing documents on the NRC docket and to identify specific licensing commitments.

4.3 Incorporation of Licensing Commitments into SDBDs

The Company's SDBDs specifically identify licensing basis requirements in each SDBD as follows:

- Chapter 8.0 of each SDBD specifically identifies the licensing commitments that are applicable in specific categories,
- Chapter 13.0 addresses each of the commitments identified in Chapter 8.0, as well as “self-imposed” commitments, such as compliance with a Regulatory Guide, and
- throughout other SDBD chapters as applicable.

Licensing commitments are appropriately annotated and a bibliographic reference is presented in Chapter 25.0, “References” of the SDBDs.

5.0 System Design Basis Documents

5.1 Description

A System Design Basis Document (SDBD) is a comprehensive document that establishes, summarizes, describes, and defines the specific functions, design criteria, design requirements, or other design basis information of a system whose boundary is defined in the document.

An SDBD is one of a series of documents prepared to establish and document the design bases for the nuclear power stations. Each major system installed in the plant is covered by a separate SDBD. SDBDs are prepared for safety related and non-safety related systems. Tables 10.1-1 and 10.1-2 identify the specific systems for which SDBDs are being prepared. A companion document, the Plant Design Basis Document (PDBD), defines the design basis of the overall plant, including design bases that are applicable to all systems as discussed in Section 6.0 of this Attachment.

An SDBD serves as both an encyclopedia of system design information, as well as a "road map" or "directory" to more detailed supporting information. While the SDBD is a stand alone document based on style and content, full utility can only be achieved when the reference documents are available to the user. The referenced design and engineering documents constitute the design records required by Appendix B to 10 CFR 50.

An SDBD is an assimilation and compilation of existing design information that is documented in a consistent, logical, and systematic manner. Each important "statement of fact," "Design Requirement," and "Design Feature" is annotated with a complete bibliographic reference that can be used to identify and retrieve the source document. An SDBD does not “create” or change any design basis information. The creation or modification of design bases information is accomplished by specific engineering and design control processes under the cognizance of the design authority. An SDBD simply presents design bases information in an organized, user friendly manner. The proper use of an SDBD is as a directory or road-map to specific engineering information documents, not just simply using the summary information in an SDBD without checking the source documents. The directory concept is important because it clearly identifies the specific source reference document upon which a specific requirement or feature in an SDBD is based. Since an SDBD is only updated periodically, this method provides assurance that a user confirms the information before use, as well as providing much more detailed information which is usually required to fully evaluate a condition or to design a plant modification.

The SDBD is a controlled configuration document and is maintained correct and reasonably up to date by the design authority. Although the design information contained in the SDBD is intended to be accurate and authoritative, occasions may arise in which other documents or physical realities appear to be, or are, in conflict with the SDBD information. These situations are referred to the design authority for resolution via the "SDBD Change Request." The design and engineering documents required per Appendix B to 10 CFR 50 have precedence over information in an SDBD.

The SDBD is prepared to serve a number of users. While the primary users are designers, design engineers, discipline engineers, system engineers, and project engineers, the SDBD has significant value to other Company personnel including, but not limited to, operating, licensing, maintenance, and training personnel.

5.2 SDBD Content and Organization

An SDBD consists of twenty-five (25) chapters as outlined below:

Chapter 1.0, "INTRODUCTION AND GENERAL INFORMATION," presents fundamental information about the SDBD, including its organization, purpose, scope, association with other documents, abbreviations and definitions, unresolved issues, etc. This chapter is essentially the same for all SDBDs and uses standard "boilerplate" wording, customized for the specific station and system that the SDBD covers.

Chapter 2.0, "SYSTEM FUNCTIONS AND PERFORMANCE CRITERIA," defines the specific functions of the system, including safety related functions (SR), non-safety related functions with special regulatory significance (NSQ), and non-safety functions (NS). Performance criteria for each of the functions are summarized. The chapter also identifies the standard ANSI N45.2.11 design topics applicable to the system.

Chapter 3.0, "SYSTEM DESCRIPTION," provides the user with a fundamental understanding of the system layout and operation under various plant operating modes. A summary of major components is tabulated. While this chapter does not contain any design basis information per se, it does provide an overview of the system and simplified diagrams to familiarize the user with the subject system's arrangement and operation.

Chapter 4.0, "SYSTEM CLASSIFICATION AND BOUNDARY," identifies the classification of the system based on Company, Nuclear Regulatory Commission (NRC), and industry classification schemes. This chapter defines the precise boundaries for the system, including mechanical (process), electrical, instrumentation and control, and building/structural boundaries.

Chapter 5.0, "SYSTEM INTERFACES," identifies the interfaces of the subject system with other "supporting" and "supported" systems. Location and arrangement interfaces are presented.

Chapter 6.0, "SYSTEM PARAMETERS," identifies and summarizes the key system parameters that are established to satisfy the plant safety analysis and the design requirements of the system. Tabular presentation of the information simplifies its use.

Chapter 7.0, "DESIGN BASIS BACKGROUND," describes the design philosophy of the system and the historical evolution of system specific issues or requirements.

Chapter 8.0, "REGULATORY BASIS AND REQUIREMENTS," identifies the regulatory requirements that govern the design of the system. Compliance with regulatory guidance documents, such as Regulatory Guides, is specifically addressed. Requirements of the NRC operating license and other licensing commitments related to the system are identified and addressed.

Chapter 9.0, "INDUSTRY CODES AND STANDARDS," identifies the specific industry codes and standards that support the design basis. Applicability of specific editions, versions, or revisions to the codes and standards is addressed.

Chapter 10.0, "DESIGNER STANDARDS AND GUIDELINES," identifies the internal standards and guidelines of the architect/engineer (A/E) and the Nuclear Steam Supply System (NSSS) manufacturer that are applicable to the design basis. These non-industry standards and guidelines were typically generated based on individual company research and design philosophy and are usually applicable to a specific type of plant, NSSS, etc. During the early days of nuclear power, many of the current-day codes and standards did not exist, therefore, the designers' judgment was used in establishing much of the design basis.

Chapter 11.0, "SAFETY ANALYSES," addresses the system-specific safety analyses that constitute the licensing basis of the plant. The plant safety analyses constitute a fundamental design input to the system design basis. The four ANSI N18.2 event categories are identified and discussed. Of particular interest are the assumptions, inputs, and results of the safety analyses since they must be implemented by the system design.

Chapter 12.0, "MARGIN EVALUATION," identifies the important safety margins that exist in the system design. These margins provide the basis for a number of evaluations required by the regulations and often provide the justification for continued unit operation when nonconforming conditions are identified.

Chapter 13.0, "SYSTEM SAFETY DESIGN CRITERIA," identifies the safety related criteria that the system must satisfy. These criteria are derived directly from the system functions (Chapter 2.0) and regulatory requirements (Chapter 8.0). In general, this chapter addresses compliance with each applicable General Design Criterion (GDC), other specific regulations, and licensing basis commitments.

Chapter 14.0, "DESIGN REQUIREMENTS," identifies and discusses the design requirements of the system and components that are implemented to satisfy the regulatory requirements, criteria, codes, and standards that have been identified. This chapter specifically addresses the design input requirements contained in ANSI N45.2.11. Each ANSI topic (28 topics) is addressed for the system and for each major component.

Chapter 15.0, "RELEVANT DESIGN ISSUES," addresses other important "programmatic" and "regulatory" issues that must be considered in the design basis. Although the design issues presented in this chapter may be considered functional design requirements as addressed in Chapter 14.0, they are included as a separate chapter because of their regulatory significance and the specific emphasis and importance that has historically been placed on the issues to facilitate continued compliance.

Chapter 16.0, "CRITICAL CALCULATION SUMMARIES," contains summaries of the critical process calculations applicable to the system. Key safety related calculations for the parameters that are identified in Chapter 6.0 (SYSTEM PARAMETERS) are presented. For systems that are not safety related, critical calculations are considered to be important process calculations that support the key parameters.

Chapter 17.0, "DESIGN BASIS MAINTENANCE REQUIREMENTS," identifies fundamental maintenance requirements that are applicable to the system or system components. While these requirements do not constitute a formal part of the design basis, identification of maintenance requirements herein serves to highlight key items. Inadequate or incorrect maintenance can have a significant effect on the design basis and justifies inclusion. The requirements presented in this chapter are directed toward the function of a component and do not address specific manufacturer's requirements or recommendations. This chapter identifies maintenance requirements needed to assure the continued validity of the design basis assumptions.

Chapter 18.0, "DESIGN BASIS COMPONENT PROCUREMENT REQUIREMENTS," specifies the fundamental requirements applicable to system components. This chapter does not contain design basis requirements, but merely summarizes and reiterates requirements presented in the SDBD.

Chapter 19.0, "DESIGN BASIS SURVEILLANCE REQUIREMENTS," identifies the surveillance requirements and their bases that must be implemented to assure the continued validity of the design basis.

Chapter 20.0, "MODIFICATION HISTORY," presents a history of modifications made to the system. The modifications are presented chronologically by Design Change Package (DCP) number, including a brief description of the change, purpose, the effect on the design basis, and the effect on margins. Significant Engineering Work Requests (EWRs) that affect the system are also identified (EWRs are no longer used to modify the plants).

Chapter 21.0, "TECHNICAL SPECIFICATION BASES," identifies items to be included in the Technical Specification. The plant safety analyses and design bases provide the basis for the Technical Specifications.

Chapter 22.0, "DESIGN BASIS OPERATIONAL REQUIREMENTS," identifies fundamental operational requirements or functions that are relevant to the design basis, including such items as human operator response time, human factors, valve throttling, etc.

Chapter 23.0, "SETPOINT BASIS," identifies the basis of key safety-related setpoints, including any applicable precautions and limitations.

Chapter 24.0, "OPEN ITEMS," identifies any open items or unresolved issues, including errors and omissions identified in preparing the SDBD.

Chapter 25.0, "REFERENCES," contains a bibliography of all references identified in the SDBD. Each reference is assigned a unique identifier that is used in the body of the SDBD. The reference number is unique to the SDBD in which it appears. References included in the DBDLS are identified with the DBDLS Control Number.

6.0 Plant Design Basis Document

6.1 Description

The engineering design bases of each nuclear plant is documented in two types of documents, a Plant Design Basis Document (PDBD) and the System Design Basis Documents (SDBDs) discussed above. A single PDBD is prepared for each nuclear station and multiple SDBDs are prepared for major systems that comprise the plant. The fundamental concept that determines whether design information appears in the PDBD or SDBDs is the uniqueness and scope of application of the design information to be addressed. Design information that is applicable to the station or multiple systems is presented in the PDBD, whereas, unique system requirements are presented in SDBDs. References to the PDBD are included in SDBDs where appropriate. Normally, information contained in the PDBD is not repeated in the SDBD.

The PDBD is defined as a comprehensive document that:

- establishes, summarizes, and defines the specific design criteria and design requirements of the nuclear facility, including those features applicable to the site, buildings, and structures,
- addresses features that are applicable to multiple systems such as seismic design, cabling, piping, etc.,
- addresses features that are common to the site or to all systems, and
- identifies the plant safety analysis design bases.

The fundamental advantage of this two-tier design basis document approach - a PDBD and multiple SDBDs per plant - is the consolidation of common design basis information within a single document (PDBD) that does not have to be extensively duplicated in other applicable SDBDs.

6.1.1 Purpose

The overall goal of the PDBD is to provide a single document that provides a summary of the design bases requirements for topics that are not system specific. Like SDBDs, the PDBD is a directory or roadmap to the design and engineering information required by Appendix B to 10 CFR 50 and ANSI N45.2.11. The PDBD merely compiles, assimilates, and organizes design basis information in such a manner as to make it more useful and accessible and does not supplant the official design and engineering documents.

The basic objectives of the PDBD are to document the fundamental:

- design criteria of the nuclear facility, i.e., site, buildings, and structures that are common to plant systems,
- topical design criteria for plant systems,
- nuclear safety requirements for the nuclear station, i.e., plant safety analyses of record, and
- design bases regulatory requirements for the nuclear facility.

6.1.2 Scope

A PDBD is prepared for each nuclear station (North Anna and Surry Power Stations). The specific scope of a PDBD includes those fundamental design basis criteria identified below. The PDBD is intended to be a summary of fundamental design basis criteria and to serve as a directory to supporting design and engineering information.

6.2 PDBD Organization and Content

The PDBD is divided into major subsections, the contents of which are summarized below.

6.2.1 Introduction

The PDBD "Introduction" provides general information about the PDBD and the nuclear power station. In addition to the routine administrative treatment of the PDBD contents similar to the SDBD, it addresses:

- identification and description of the facility,
- basic information about the facility, such as location, coordinates, exclusion area, site property boundaries, etc.,
- basic information about the generating units (e.g., type, thermal rating, NSSS, AE, etc.),
- the PDBD in relation to SDBDs, and
- the PDBD in relation to the UFSAR.

The "Introduction" may contain other general information that may be useful in orienting the user to the facility and the PDBD itself, such as site maps, general licensing history, etc.

6.2.2 Plant Safety Analyses

The plant safety analyses demonstrate that the plant can be operated safely and in accordance with the criteria established by the NRC regulations. Since plant safety analyses may change over the period of the facility license, it is important that the analyses of record included in the plant's current licensing basis be identified. In conjunction with the NRC regulations themselves, the Plant Safety Analyses constitute the most important design bases for the nuclear generating units. The assumptions and results of safety analyses establish the "functions" and "parameter values" of accident mitigation systems and components, as well as establishing the reference bounds of design for major systems and components. Indeed, it is these functions and limiting values that constitute the design bases as defined by 10 CFR 50.2.

The PDBD identifies the current Plant Safety Analyses of Record that is part of the current licensing basis. This section of the PDBD identifies each safety analysis by event category and provides a summary of the event, assumptions, results, etc. Since plant safety analyses are so fundamental to the design bases of plant safety systems addressed in PDBDs, significant emphasis has been placed on this section of the PDBD.

6.2.3 Regulatory and Nuclear Safety Criteria

The fundamental design criteria of a nuclear generating station/unit are established by NRC rules and regulations as identified in Title 10 of the Code of Federal Regulations. In addition to the basic requirements contained in the regulations, there are many NRC guidelines and industry standards that may be invoked by the licensing basis of the plant.

There are a number of nuclear safety criteria and functional design criteria for the nuclear facility, including site, structure, systems, and components, contained in the regulations, NRC guidance documents, the licensing basis, and referenced codes and standards. While it is not the intent to identify every regulation, guideline, industry code or standard, and licensing basis commitment that is applicable to a plant in the PDBD, the PDBD does address the fundamental framework that constitutes the nuclear safety criteria applicable to a specific plant. A fundamental goal of the PDBD is to provide a perspective to the integration of a number of criteria to ensure that specified nuclear safety criteria are identified so that the design authority may properly implement such criteria.

The PDBD, in conjunction with the SDBDs, addresses the important nuclear safety criteria. The regulatory requirements are contained in the plant's licensing basis, which is addressed on a case basis in the PDBD and SDBDs.

The General Design Criteria (GDC), Appendix A to 10 CFR 50, or those invoked by the plant's licensing basis are identified in this section.

6.2.4 Site Design Basis Conditions

This section of the PDBD addresses those site conditions that constitute the fundamental engineering design bases of plant structures, systems, and components. Site conditions significantly affect the design bases of structures, systems, and components and constitute a group of criteria that are applicable to nuclear safety. The site conditions are unique to each plant and they must be evaluated and considered as part of the engineering design bases to ensure that safety functions can be accomplished over the range of conditions that might prevail at the site.

This section of the PDBD also addresses the range of external phenomena, natural, and man-made hazards, that might exist at the plant site and that constitute engineering design basis requirements for the plant structures, systems, and components. The objective of this section is to identify those site conditions or parameters, natural or man-made, that establish fundamental engineering design basis requirements for plant structures, systems, and components.

6.2.5 Buildings and Structures

This section of the PDBD addresses the engineering design bases of plant buildings and structures, with emphasis on those that are important to safety. The approach followed to identify the engineering design bases of plant structures and buildings is to identify the individual structures and buildings for which design requirements are specified, and for each structure or building, identify the applicable design parameters and design criteria. This information is presented in an orderly, systematic manner, normally a table for each structure and building identifying the appropriate parameters. The structures and buildings are keyed to a site plan or other appropriate drawing.

6.2.6 Plant Systems

This section of the PDBD provides a summary and overview of the major systems that comprise the plant, providing an objective framework upon which to establish and define the engineering design bases of such systems, including the integration of systems to accomplish the nuclear safety functions. The identification of specific boundaries of each system is prerequisite to defining the engineering design bases of a system and the system interfaces. The system classification scheme is addressed. For those systems for which an SDBD is not developed, this section identifies and summarizes pertinent information about the system. Thus, the PDBD provides complete coverage of the plant systems.

6.2.7 Engineering Programs

This section identifies the significant engineering programs maintained by the design authority. Engineering programs are normally very structured, have an engineering sponsor, and are maintained on a continuous basis. These engineering programs and their bases are usually well defined and documented, therefore, only summary information is provided in the PDBD and the engineering program or program document is referenced.

6.2.8 Topical Design Criteria

This section of the PDBD identifies the nuclear design criteria that are applicable to multiple systems. The topical design criteria are addressed in this section and are referenced by individual SDBDs. For purposes of presentation, the topics are organized by engineering discipline and are identified below. Since the topics are "standard" within the nuclear industry and are reasonably self-explanatory, no discussion of each topic is presented herein. The following are typical topics that may be included in the PDBD:

Electrical Topical Design Criteria

- electrical system coordination and protection,
- electrical system analysis (voltage, current, and short circuit),
- electrical breaker, fuse, and starter design criteria,
- grounding and lightning protection,
- single failure criterion,
- electrical separation and diversity,
- cable and cable sizing design criteria,
- cable trays, conduit, and raceway design criteria,
- electrical motor design criteria,
- containment electrical penetration criteria,
- isolation of electrical equipment,
- cathodic protection,
- electrical equipment qualification, and
- station blackout design criteria.

Mechanical Topical Design Criteria

- high energy line break design criteria,
- single failure criterion,
- containment mechanical penetration criteria,
- equipment loading criteria,
- piping design criteria,
- coatings and painting criteria, and
- heavy loads.

Nuclear Topical Design Criteria

- nuclear core and internals design,
- radiological shielding criteria,
- radiation protection design criteria,
- radiation source terms criteria,
- as low as reasonably achievable (ALARA) criteria, and

- primary containment analysis criteria.

Instrumentation/Control Topical Design Criteria

- single failure criterion,
- independence and separation criteria,
- cable and wiring criteria,
- setpoints and uncertainty criteria, and
- control room design criteria.

Civil/Structural/Engineering Mechanics Topical Design Criteria

- seismic design criteria,
- equipment foundation and support design criteria,
- general structural design criteria,
- structural attachments design criteria,
- cranes/lifting devices,
- openings and penetration in buildings design criteria,
- missile protection (internal and external) design criteria,
- flooding analyses design criteria,
- piping supports design criteria,
- containment penetration criteria,
- seismic qualification of equipment design criteria, and
- missile design criteria.

General Topical Design Criteria

- single failure design criterion,
- containment isolation design criteria,
- liquid, solid, and gaseous radioactive waste design criteria,
- control room habitability design criteria,
- human factors design criteria,
- auxiliary shutdown capability design criteria, and
- plant hazards analysis.

7.0 Open Item Management

7.1 Introduction

This section describes the process for managing and addressing open items that are identified during the DBD development process. Inherent in the implementation of a DBD Program is the identification of open items that may include questions, concerns, missing information, or other discrepancies. A logical systematic approach to managing and dispositioning open items identified during DBD development has been established to assure that open items are reviewed and evaluated for safety significance. Appropriate categorization and prioritization criteria have been established and applied to open items. The process for dispositioning open items is addressed herein. As used herein, DBD refers to both the PDBD and the SDBDs.

Open items for each SDBD are identified in Chapter 24.0 of each SDBD. Open items in the PDBD are included in a numbered section. Open items include a broad range of items that may be characterized as:

- missing source documentation,
- referencing a “secondary” reference when a “primary” reference is preferred,
- conflicting information or data in referenced documents,
- potential errors or discrepancies in calculations, reports, or other documents that may be identified in the course of DBD development,
- incomplete supporting information,
- missing or incomplete calculations,
- inconsistency with regulatory requirements or commitments such as the UFSAR, Technical Specification, correspondence, etc.,
- other matters judged to require additional review or action, and
- linkages to other SDBDs or PDBD that have not been developed.

7.2 Identification of Open Items

Open items are identified throughout the development cycle of a DBD. The development of a DBD is a comprehensive and complex process. Since the process is primarily a compilation and assimilation of existing engineering information and documents, most open items are associated with the availability, completeness, accuracy, or quality of documents or information referenced in the DBD. As the development process proceeds, each individual developer is responsible for maintaining a running list of open items for possible inclusion in the issued DBD. Experience has shown that as development proceeds some open items are resolved and removed from the list. During the development of a DBD, open items are tracked and dispositioned by the DBD developer. When a "draft" DBD is completed, a list of remaining open items is contained therein.

The DBD Program uses the Potential Problem Report (PPR) Process discussed in response to Item (d) of the NRC letter to identify potential safety concerns. The PPR process is used to assure that potential safety concerns are reviewed and addressed using the Company corrective action process when applicable. In the event the Potential Problem Report process determines that a deviation report is required, the related open item is reviewed for operability and reportability as discussed in response to Item (d) of the NRC letter.

After a "draft" SDBD is completed, it receives relatively widespread distribution, including station and corporate design authority personnel. This review process makes the open items available to others to identify any potential safety concerns that may not have been recognized by the developers or the DBD staff. Individual reviewers of the SDBDs may make their own determination regarding the safety significance of an open item and initiate a "Potential Problem Report" or "Deviation Report" as appropriate.

7.3 Disposition of Open Items

The process for dispositioning open items is outlined below:

- a DBD staff member reviews each open item and assigns it to the appropriate organization for disposition,
- if an open item does not require immediate resolution, a basis of deferral is required to indicate that there are no significant safety implications associated with the deferral of the open item,
- a strategy plan for dispositioning each open item is established,
- the status of the open item is tracked until it is dispositioned,
- when an item is dispositioned and closed, it is identified for deletion in a forthcoming DBD revision, and
- when the DBD is revised, open items that have been closed are deleted and appropriate changes are made to the body of the SDBD.

Responsibility for dispositioning open items is assigned to those organizations that are sponsors of the engineering programs or design information that may be associated with the open item, or that are otherwise qualified and suited to address the open item. For open items that affect an engineering program or activity that normally would require the initiation of a "change request," the responsible organization is required to review the open item to verify that a "change request" is appropriate. If appropriate, a "change request" is required to be completed in accordance with applicable Company procedures.

Within approximately 90 days of SDBD issuance, an Open Item Review Meeting (OIRM) is conducted for each Group 2 SDBD and later SDBDs. The OIRM concept was not originally invoked for Group 1 SDBDs however OIRMs were retroactively conducted for the Group 1 SDBDs based on commitments made during NRC discussions regarding open items. The purpose of the OIRM is to discuss the open items, their assignment, basis for deferral, and other relevant concerns.

When an open item is dispositioned, the assignee notifies the DBD staff. The DBD staff reviews the disposition and determines its acceptability. The disposition of the open item is typically incorporated into a DBD during the next revision. The revision includes the deletion of the open item from the DBD and revision of the text of the DBD to reflect the disposition, as appropriate.

8.0 DBD Control And Maintenance

8.1 Issuance and Control

The distribution of DBDs is accomplished by issuing hard copies (loose-leaf binders) to assigned personnel via the Company's Records Management process. The DBDs are identified in the Document Management Information System (DMIS) and are available at multiple locations. This provides broad access to issued DBDs. The DBDs may be issued to engineering group libraries or to an individual. Records Management personnel maintain the controlled distribution list for DBDs. The DBDs are issued and controlled in accordance with applicable Company procedures.

8.2 Revisions and Updating

To maximize the utility and reliability of DBDs they should be complete, accurate, and reasonably current. Therefore, an appropriate DBD revision process must be administered.

The target frequency of revising an SDBD is currently every 12 to 18 months, but this objective has not been achieved to date. The Group 1 SDBDs (18 total) have been revised once. Except for the two (2) most recently issued SDBDs (January 1996), the other SDBDs have not been revised within the target frequency.

The fundamental reason that justifies any revision frequency that is not "real time" is that the DBDs are not the "design and engineering documents" that are required by Appendix B to 10 CFR 50 and ANSI N45.2.9, but serve as a summary compilation of design and engineering information and a directory to that information. It is the actual "official" design and engineering documents that must reflect the most current conditions and configuration. Users of the DBDs for design control activities must consult and use the actual design and engineering documents, including the appropriate revision of such documents, not just DBDs.

The currentness issue of the SDBD has been identified as a potential concern and must be evaluated to determine the improvements that are reasonable. However, proper use of the DBDs provides assurance that the design and engineering documents required by Appendix B to 10 CFR 50 and ANSI N45.2.11 are reviewed to evaluate a condition or develop a design change.

9.0 Related Engineering Programs And Support Activities

Heretofore, this document has focused on the Design Basis Document Program, primarily as related to developing SDBDs and PDBDs. While DBDs are the products produced by the DBD Program, providing a compilation of, and directory to, the design and engineering information that supports the design bases, the actual engineering documents themselves constitute the official engineering design bases of the plant. The design authority establishes and maintains the appropriate documentation to satisfy Appendix B to 10 CFR 50 and ANSI N45.2.11 - the DBDs provide a roadmap of this information. During the DBD development effort, enhancements have been made, or are being made, to some of the engineering programs and related documents. This section identifies and discusses some of the more important enhancements made in conjunction with the DBD development effort. This section is not a complete list of engineering programs, nor a complete compilation of all enhancements that have been made, but merely demonstrates that DBD development has driven, or benefited from, engineering program improvements during this period.

9.1 Plant Safety Analyses

The Plant Safety Analyses are probably the most important design and engineering information that are established to analyze postulated events and assure that the plant can operate safely. At the onset of the DBD program, the importance of the Plant Safety Analyses was identified and appropriate safety analysis information is included in each SDBD as related to the subject system. The Plant Safety Analyses establish the most important design bases as defined by 10 CFR 50.2, since they identify the important safety functions required for accident mitigation and establish key parameter values that establish the reference bounds of design. Indeed, the Plant Safety Analyses constitute the basis of most of the safety limits, limiting safety system settings, and limiting conditions of operation as contained in the Technical Specifications. A reasonable interpretation of design bases per 10 CFR 50.2 is that the "functions" and "values" constituting the reference bounds of design identified in this regulation are the Technical Specification functions and values. The Plant Safety Analyses determine most of the design requirements of safety related systems and components, therefore, the safety analyses are very important.

In addition to the inclusion of information within each SDBD, as applicable, a major effort was initiated to enhance the documentation of the Plant Safety Analyses by developing an Accident Analysis Design Basis Document (AADBD) for each station. The purpose of an AADBD is to consolidate the assumptions of design basis accident analyses that support the safe operation of the nuclear units, and to clearly identify the licensing basis analyses of record for each acceptance criterion of each UFSAR accident.

The AADBD program was initiated in 1991 and an AADBD has been prepared for both North Anna and Surry Power Stations. The AADBDs have been substantially completed for the UFSAR accident analyses, except for the small and large break Loss-of-Coolant Accident which are currently under development. It is anticipated that the AADBDs, including those accidents identified above, will be completed during 1997.

9.2 Equipment Classification List (Q-List) Program

The Equipment Classification List Program, commonly referred to as Q-List, is a program that primarily develops and maintains a listing of components based on the Company's equipment classification scheme to ensure that appropriate quality levels are established. The Q-List is an integral part of configuration control. Equipment is classified as 1) safety related, 2) non-safety related with special regulatory significance, and 3) not safety related. The primary objective of the Q-List is to clearly identify the classification of significant components, including the basis of the classification. This program was initiated in the mid-1980's in response to NRC Generic Letter 83-28 and includes three phases.

Phase 1 established quality classifications for unclassified mark numbers created during the Physical Verification Project described in response to Item (c) of the NRC letter. The classifications were based on drawing reviews and conservative classification criteria. Phase 2 activities established a Quality Classification Analysis functional basis for the mark numbers classified as safety related in Phase 1. Phase 3 is ongoing and addresses validating the Q-list based on a review of mark numbers listed in the Equipment Data System against the components identified on station drawings. This validation was initiated on a system-by-system basis and is documented in Quality Classification Analyses for each safety related and non-safety related with special regulatory significance mark number.

Each SDBD identifies the classification and function of a system and provides significant information about major components. The system functions as identified in an SDBD provide the fundamental basis for classifying components.

9.3 Environmental Qualification Program

The Environmental Qualification (EQ) Program was established and implemented in accordance with 10 CFR 50.49 before the initiation of the DBD Program. The EQ Program addresses safety related electrical equipment located in a harsh environment and relied upon during and following accidents. The Environmental Qualification Improvement Effort was initiated by Virginia Power to confirm implementation of 10 CFR 50.49. A detailed technical review was performed to confirm the accuracy, ease of use, consistency and completeness of environmental qualification documentation, the Environmental Qualification Master List, and Environmental Zone Descriptions. Outstanding changes were incorporated into the Qualification Documentation Review Packages. These packages document the bases for equipment qualification. Program activities were prioritized to first address equipment with limited qualified life or regular procurement activities, followed by equipment with high usage.

All activities are complete. The Qualification Documentation Review Packages for some electrical cable were previously established and considered adequate. This was confirmed by NRC inspections.

9.4 Fire Protection Program

The Fire Protection Program required by 10 CFR 50.48 and Appendix R to 10 CFR 50 was established and implemented prior to the initiation of the DBD Program. However, each SDBD, as applicable, addresses the fire protection requirements relative to the Appendix R (10 CFR 50) requirements. In general, an SDBD directs the user to the "Appendix R Report" which is the official controlled document that documents compliance with Appendix R to 10 CFR 50. An SDBD for the Fire Protection System has also been prepared for each station. This SDBD addresses the fire protection system including fire detection, fire suppression, Appendix R, etc., in standard SDBD format.

The Appendix R Program was enhanced to confirm implementation of 10 CFR 50, Appendix R requirements. The program addresses the systems, structures and components required for safe shutdown in the event of an Appendix R fire. A program was initiated in 1992 to enhance and refine the long-term Appendix R program and controls. The program included increased self-assessment to confirm that the

Appendix R program is addressed and that effective change mechanisms exist for tracking Appendix R recommendations and action plans.

In addition, a Fire Barrier Penetration Seal Inspection and Repair Program was initiated at Surry to inspect, repair when needed, and document the condition of each individual fire barrier electrical penetration seal. A design basis verification of 10 CFR 50, Appendix R emergency lighting was also initiated.

In summary, the review and validation efforts associated with Appendix R to provide assurance that the design basis requirements relative to fire protection are incorporated into the physical plant and associated documents have provided additional validation of the design bases relative to fire protection.

9.5 Post Accident Monitoring Verification Program (Regulatory Guide 1.97)

In accordance with Company commitments relative to Regulatory Guide (RG) 1.97, an engineering program was established and implemented prior to the initiation of the DBD Program. However, each SDBD, as applicable, addresses the RG 1.97 requirements relative to the post accident monitoring of certain parameters.

The Regulatory Guide 1.97 Verification Program was initiated by the technical program owners to confirm implementation of the requirements provided by Table 1 (Design and Qualification Criteria For Instrumentation) and Table 3 (PWR Variables) of NRC Regulatory Guide 1.97. A technical review was performed to confirm the completeness of the components identified as PWR variables and to confirm adherence of these components to the guide's Design and Qualification criteria. A line-by-line comparison between the Regulatory Guide's PWR Variables (Table 3) and actual installed plant instrumentation loops was done to determine the instrumentation loops that should be identified as being associated with Regulatory Guide 1.97 requirements. The program then evaluated each variable from "sensor to display" against the Regulatory Guide Design and Qualification Criteria (Table 1). Where applicable components were modified or upgraded to meet the criteria of Table 1 or exceptions were taken as outlined in correspondence between Virginia Power and the NRC. This effort is complete.

In summary, during the last several years significant effort has been made to enhance the design basis documentation relative to RG 1.97 which has served as a validation effort for this important design topic.

9.6 Seismic Adequacy of Components

A seismic qualification verification program was initiated to assess implementation of the requirements of 10 CFR 50, Appendix A, Criterion 2 as well as 10 CFR 100, Appendix A. Enhancements were made to the seismic qualification program for safety related and safety significant electrical and mechanical equipment. A detailed review of safe shutdown equipment is being conducted, including walkdowns and evaluations that rely on analysis, test data, and earthquake experience data. This review has enhanced the seismic capacity of several components through physical modification.

Examples include connecting adjacent cabinets containing essential relays, strengthening equipment anchorage and structural load paths, and modifying conditions and/or components to alleviate seismic interaction concerns. Seismic evaluation packages are being prepared to document the basis of seismic verification.

For seismic qualification of new and replacement equipment, engineering standards, procedures, and technical reports have been developed. The previous design bases are enhanced where possible by conforming to IEEE-344-1975 and NRC Regulatory Guide 1.100, Revision 1 for seismic qualification of electrical equipment. Where applicable, and on a case-by-case and systematic basis, earthquake experience data and previous seismic test data is also applied to new and replacement equipment.

Initial plant walkdowns have been completed and the evaluations that demonstrate seismic adequacy are in progress and are approximately 60% complete. The current schedule for submitting our summary reports is May 1997 for North Anna Power Station and November 1997 for Surry Power Station.

In summary, this seismic design validation effort provides added assurance that the seismic design basis requirements for components have been appropriately reflected in the physical plant.

9.7 Other Support Programs

The aforementioned list of programs is not intended to be a complete list of engineering programs, but has identified those programs that are related to the DBD Program, programs addressed by open items in the SDBDs, and programs that have been enhanced during the time frame of the DBD Program. Since a DBD is comprehensive, other programs may have been affected by DBD development or DBD open items. While not specifically mentioned herein, open items related to these programs are processed like other open items. As appropriate, the open items are dispositioned and the program documentation is revised accordingly.

10.0 Status And Schedule

The previous sections have described the Company's DBD Program. This section summarizes the status of the development activities and presents a schedule for completion of the program.

10.1 SDBD Development

Currently, 41 SDBDs have been completed for both stations (21 for Surry and 20 for North Anna). Tables 10.1-1 and 10.1-2 tabulate the SDBDs that have been issued for North Anna and Surry, respectively, the SDBDs that are currently being developed (working), and those SDBDs that are currently planned. In these tables, the SDBDs are presented in Groups 1 through 8. These groupings are for administrative and contract purposes and have no significance otherwise.

The systems for which SDBDs are being developed are substantially fixed since most have been developed; however, other SDBDs may be developed if the need is identified. The estimated completion dates are targets only, but it is planned to issue SDBDs by June 30, 1999.

A significant number of open items have been identified during the DBD development activities performed to date. For the most part, these open items have not yet been closed, however, the significance of the open items is assessed during dispositioning.

10.2 SDBD Revision

Tables 10.1-1 and 10.1-2 also indicate the most current revision of each SDBD. As used in the tables "Revision 00" means the originally issued SDBD, "Revision 01" means the first revision, and "Revision C" means Draft C. A Draft C SDBD is one for which development has essentially been completed, and is ready for final review and issuance. These tables indicate that 18 of 41 issued SDBDs have been revised since their original issuance.

After all SDBDs have been issued, the Company's current plan is to revise and reissue each SDBD approximately every 12 to 18 months. The revision frequency will be evaluated to determine if more frequent revisions are needed.

10.3 PDBD Development

10.3.1 Status

The AADBD is used to generate the plant safety analysis portion of the PDBD. The AADBD for each station is approximately 90% completed. Work is in progress for other sections of the PDBD, but is not complete.

10.3.2 Schedule

The PDBD for each station is scheduled for issuance by approximately June 30, 1999.

10.4 PDBD Revision

10.4.1 Status

The PDBD has not been issued, therefore, it is not ready for revision.

The AADBD is being maintained current as part of the safety analysis engineering process.

10.4.2 Schedule

When issued, it is currently planned that the PDBD will be revised approximately every 12-18 months.

Table 10.1-1
SUMMARY STATUS AND DEVELOPMENT SCHEDULE FOR SYSTEM DESIGN BASIS DOCUMENTS (SDBD) NORTH ANNA POWER STATIONS - UNIT NOS. 1 AND 2 VIRGINIA ELECTRIC AND POWER COMPANY

		STATUS	REVISION	
GROUP: 01				
AFW -	Auxiliary Feedwater	Issued	01	
ED -	Emergency Power (125V DC)	Issued	01	
EG -	Emergency Diesel Generator	Issued	01	
EP -	Emergency Power	Issued	01	
IA -	Instrument and Service Air	Issued	01	
QS -	Quench Spray	Issued	01	
RS -	Recirculation Spray	Issued	01	
SI -	Safety Injection	Issued	01	
SW -	Service Water	Issued	01	
GROUP: 02				
CC -	Component Cooling Water	Issued	00	
CN -	Condensate	Issued	00	
ESS -	Station Service Power	Issued	00	
FW -	Feedwater	Issued	00	
MS -	Main Steam	Issued	00	
GROUP: 03				
CH -	Chemical and Volume Control	Working	C	EST. COMPLETION 12/98
NC -	NSSS Control System	Issued	00	
NI -	Nuclear Instrumentation	Issued	00	
RC -	Reactor Coolant	Working	C	EST. COMPLETION 12/98
RH -	Residual Heat Removal	Issued	00	
RPS -	Reactor Protection	Issued	00	
GROUP: 04				
EA -	Power Generation	Issued	00	
EV -	Vital Bus (120/240V)	Issued	00	
GROUP: 05				
CD -	Chilled Water	Working	C	EST COMPLETION 6/97
GW -	Gaseous Waste	Working	C	EST. COMPLETION 6/97
HA -	Auxiliary Building Ventilation	Working	C	EST. COMPLETION 12/97
HC -	Control Room Air Conditioning and Pressurization	Working	C	EST. COMPLETION 12/97
HL -	Service Building Ventilation	Working	C	EST. COMPLETION 12/97
HO -	Miscellaneous Building Ventilation	Working	C	EST. COMPLETION 12/97
HP -	Fuel and Decontamination Building Ventilation	Working	C	EST. COMPLETION 12/97
HR -	Containment Air Cooling	Working	C	EST. COMPLETION 12/97
HT -	Turbine Building Ventilation	Working	C	EST. COMPLETION 12/97
GROUP: 06				
CW -	Circulating Water and Vacuum Priming	Working	C	EST. COMPLETION 6/97
FP -	Fire Protection	Working	C	EST. COMPLETION 6/97
RM -	Radiation Monitoring	Working	C	EST. COMPLETION 6/97
GROUP: 07				
BC -	Bearing Cooling	Working	C	EST. COMPLETION 4/98
BD -	Steam Generator Blowdown, Recirculation and Transfer	Working	C	EST. COMPLETION 3/98
BR -	Boron Recovery and Primary Grade Water	Working	C	EST. COMPLETION 9/98
CP -	Condensate Polishing	Working	C	EST. COMPLETION 9/98
CV -	Containment Vacuum and Leakage Monitoring	Working	C	EST. COMPLETION 9/98
EC -	Communication Systems	Working	C	EST. COMPLETION 12/98
GN -	Primary and Secondary Plant Gas Supplies	Working	C	EST. COMPLETION 6/98
PHC -	Post-accident Monitoring and Hydrogen Removal	Working	C	EST. COMPLETION 6/98
PVD -	Primary Vents and Drains - Radioactive	Working	C	EST. COMPLETION 6/98
SS -	Sampling System	Working	C	EST. COMPLETION 6/98
SVD -	Secondary Drains	Working	C	EST. COMPLETION 8/98
GROUP: 08				
EI -	Electrical Instrumentation and Plant Computer	Planned		EST. COMPLETION 6/99
EL -	Station Lighting	Planned		EST. COMPLETION 6/99
FC -	Fuel Pool Cooling and Reactor Cavity Purification	Working	C	EST. COMPLETION 6/99

Table 10.1-2
SUMMARY STATUS AND DEVELOPMENT SCHEDULE FOR SYSTEM DESIGN BASIS DOCUMENTS (SDBD) SURRY POWER STATION UNIT NOS. 1 AND 2 VIRGINIA ELECTRIC AND POWER COMPANY

		STATUS	REVISION	
GROUP: 01				
AFW -	Auxiliary Feedwater	Issued	01	
ED -	Emergency Power (125V DC)	Issued	01	
EG -	Emergency Diesel Generator	Issued	01	
EP -	Emergency Power	Issued	01	
IA -	Instrument and Service Air	Issued	01	
CS -	Containment Spray	Issued	01	
RS -	Recirculation Spray	Issued	01	
SI -	Safety Injection	Issued	01	
SW -	Service Water	Issued	01	
GROUP: 02				
CC -	Component Cooling Water	Issued	00	
CN -	Condensate	Issued	00	
ESS -	Station Service Power	Issued	00	
FW -	Feedwater	Issued	00	
MS -	Main Steam	Issued	00	
GROUP: 03				
CH -	Chemical and Volume Control	Working	C	EST. COMPLETION 12/98
NC -	NSSS Control System	Issued	00	
NI -	Nuclear Instrumentation	Issued	00	
RC -	Reactor Coolant	Working	C	EST. COMPLETION 12/98
RH -	Residual Heat Removal	Issued	00	
RPS -	Reactor Protection	Issued	00	
GROUP: 04				
EA -	Power Generation	Issued	00	
EB -	Black Battery (125V DC)	Issued	00	
EV -	Vital Bus (120/240V)	Issued	00	
GROUP: 05				
CD -	Chilled Water	Working	C	EST. COMPLETION 6/97
GW -	Gaseous Waste	Working	C	EST. COMPLETION 6/97
HA -	Auxiliary Building Ventilation	Working	C	EST. COMPLETION 12/97
HC -	Control Room Air Conditioning and Pressurization	Working	C	EST. COMPLETION 12/97
HL -	Service Building Ventilation	Working	C	EST. COMPLETION 12/97
HO -	Miscellaneous Building Ventilation	Working	C	EST. COMPLETION 12/97
HP -	Fuel and Decontamination Building Ventilation	Working	C	EST. COMPLETION 12/97
HR -	Containment Air Cooling	Working	C	EST. COMPLETION 12/97
HT -	Turbine Building Ventilation	Working	C	EST. COMPLETION 12/97
GROUP: 06				
CW -	Circulating Water and Vacuum Priming	Working	C	EST. COMPLETION 6/97
FP -	Fire Protection	Working	C	EST. COMPLETION 6/97
RM -	Radiation Monitoring	Working	C	EST. COMPLETION 6/97
GROUP: 07				
BC -	Bearing Cooling	Working	C	EST. COMPLETION 4/98
BD -	Steam Generator Blowdown, Recirculation and Transfer	Working	C	EST. COMPLETION 3/98
BR -	Boron Recovery and Primary Grade Water	Working	C	EST. COMPLETION 9/98
CP -	Condensate Polishing	Working	C	EST. COMPLETION 9/98
CV -	Containment Vacuum and Leakage Monitoring	Working	C	EST. COMPLETION 9/98
EC -	Communication Systems	Working	C	EST. COMPLETION 12/98
GN -	Primary and Secondary Plant Gas Supplies	Working	C	EST. COMPLETION 6/98
PHC -	Post-accident Monitoring and Hydrogen Removal	Working	C	EST. COMPLETION 6/98
PVD -	Primary Vents and Drains - Radioactive	Working	C	EST. COMPLETION 6/98
SS -	Sampling System	Working	C	EST. COMPLETION 6/98
SVD -	Secondary Drains	Working	C	EST. COMPLETION 8/98
GROUP: 08				
EI -	Electrical Instrumentation and Plant Computer	Planned		EST. COMPLETION 6/99
EL -	Station Lighting	Planned		EST. COMPLETION 6/99
FC -	Fuel Pool Cooling and Reactor Cavity Purification	Planned		EST. COMPLETION 6/99