Seabrook Station

Response To The NRC's Letter Of October 9, 1996 On The Adequacy and Availability Of Design Bases Information

February 7, 1997

North Atlantic

The Northeast Utilities System

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The Northeast Utilities System

February 7, 1997

Docket No. 50-443
NYN-97012

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Seabrook Station
Response to the NRC's Request for Information
Pursuant to 10 CFR 50.54(f) Regarding
Adequacy and Availability of Design Bases Information

In a letter dated October 9, 1996[1], the NRC requested that North Atlantic Energy Service Corporation (North Atlantic) provide information, pursuant to 10 CFR 50.54(f), regarding the adequacy and the availability of design bases information at Seabrook Station. This letter and the attached report[2] are being submitted in response to this request.

The purpose of the NRC's letter was to obtain information that will provide added confidence and assurance that Seabrook Station is operated and maintained within the design bases and that any deviations are reconciled in a timely manner. In support of our response, we have conducted a detailed self-assessment that includes the following:

- A review of Seabrook Station's historical performance including past audits, assessments and inspections;
- Vertical Slice reviews of seven risk significant systems;
- A review of the Updated Final Safety Analysis Report (UFSAR) Chapter 15;
- A review of selected portions of the Technical Specifications;
- A review of nine important Engineering Technical Programs (e.g. Environmental Qualification, Fire Protection) and Topical Areas (e.g. Station Blackout, High Energy Line Break); and

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[1] NRC letter dated October 9, 1996, "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," James M. Taylor to Bruce D. Kenyon

[2] "Response to the NRC's Letter of October 9, 1996 on the Adequacy and Availability of Design Bases Information."

- A review of the links within and among current configuration control programs, processes and procedures.

All of the reviews conducted in support of this response have been consolidated into a North Atlantic Engineering Self-Assessment Report (ESAR). The responses to the six specific NRC requests that are contained in the attached report are based on the information contained in the ESAR.

The self-assessment provides objective evidence to support our conclusions. The Vertical Slice Reviews indicate how effectively the design bases have been maintained over the life of Seabrook Station. The accuracy and validity of the accident analyses assumptions that govern much of the design bases were examined in the UFSAR Chapter 15 review. In the Technical Specifications reviews, source documents for numerical operating limits and the appropriateness of Limiting Conditions for Operation were reviewed. Our effectiveness in incorporating the design and licensing bases into engineering program requirements was assessed in the Engineering Technical Programs and Topical Areas review. Finally, a measure was taken of the ability of our current program to ensure that the plant continues to be maintained within the design bases in the Current Programs, Processes and Procedures Review. We believe that these assessments, when considered in the context of our historical performance, provide a sufficient basis upon which to respond to the NRC's requests.

The self-assessment was the product of an intense effort that demanded a considerable commitment of personnel and resources. The reviews were conducted by knowledgeable and experienced engineers supplemented by highly qualified contractor personnel. For example, the 30-member vertical slice review team averaged over 20 years of nuclear industry experience. In total, over 15,000 person-hours were expended over a period of about two and one-half months on the project. As expected with any examination of this scope and intensity, areas warranting additional attention and improvement have been identified. Deviations and discrepancies identified by the self-assessment are being evaluated individually and collectively to ensure that we are addressing any areas of weakness that they may represent.

The self-assessment supplements our ongoing efforts to improve the configuration management and corrective action programs. As described in the report, a number of efforts have been underway in these areas. In 1995, we initiated changes to the corrective action program which include the transition to a single adverse condition reporting document, a lowering of the reporting threshold and the establishment of a management review team. Recently, we have contracted with Performance Improvement International (PII) to provide training in root cause analysis and human error reduction. PII will also certify North Atlantic staff as instructors so that this training can become an ongoing component of our corrective action program.

As a result of a detailed common cause analysis of identified UFSAR discrepancies in 1996, improvements are in the process of being implemented in UFSAR maintenance.

They include improved 10 CFR 50.59 evaluation guidance and documentation, additional training, increased oversight and the use of LAN-based software to efficiently conduct UFSAR searches.

Other initiatives that are addressing configuration management issues include the Design Basis Document (DBD) program, the procedure upgrade program and the conversion to Improved Standard Technical Specifications. North Atlantic's DBD program was established in 1988. To date, 33 DBDs have been completed. Maintenance Rule criteria will be used to establish future priorities in this important program. The procedure upgrade program, begun in 1994, will review, consolidate, simplify and clarify about 3,000 plant procedures. The conversion to Standard Technical Specifications is expected to be completed in 1998.

In addition to evaluating and dispositioning each finding and recommendation, North Atlantic is initiating a number of actions to improve performance and configuration management as a follow-up to the self-assessment. Self-assessment findings related to the UFSAR and to calculation consistency will be subjected to a common cause analysis to identify common conditions and generic causes. This will ensure that program enhancements in these areas are properly targeted and effective. A comprehensive review of the UFSAR will also be conducted. Details on the scope of this review will be submitted separately. As a result of the self-assessment, additional structure, visibility and organizational focus will be given to engineering topical areas such as High Energy Line Break and Station Blackout. Finally, North Atlantic has requested that PII conduct an independent collective analysis of the self-assessment findings. Such an analysis will provide additional assurance that we have identified opportunities for program improvement and that appropriate corrective actions are being implemented.

As Chief Nuclear Officer for North Atlantic, I am responsible for the safe operation of Seabrook Station. Based on my knowledge of the process and personnel involved in the self-assessment, the extensive oversight and review that was conducted, and my own review of the self-assessment findings, I conclude that Seabrook Station's programs, processes and procedures provide reasonable assurance that the plant is being operated and maintained within its design bases and that deviations are reconciled in a timely manner.

If you should have any questions on this submittal, please contact Mr. Terry L. Harpster, Director of Licensing (603) 773-7765.

Very truly yours,

NORTH ATLANTIC ENERGY SERVICE CORP.

Ted C Feigenbaum

Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer

cc: Mr. Hubert J. Miller, Region I Administrator
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AFFIDAVIT

STATE OF NEW HAMPSHIRE

COUNTY OF ROCKINGHAM

TED C. FEIGENBAUM, being duly sworn according to law, deposes and says as follows:

1. I am the Executive Vice President and Chief Nuclear Officer of North Atlantic Energy Service Corporation (North Atlantic). North Atlantic is authorized by an operating license issued by the U.S. Nuclear Regulatory Commission to possess, use and operate the Seabrook Nuclear Power Station.
2. I am authorized to sign and submit on behalf of North Atlantic the attached response to the U.S. Nuclear Regulatory Commission's letter dated October 9, 1996 requesting additional information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design basis information. The attached response is comprised of the preceding letter dated February 7, 1997 and the attachments thereto (the "Response").
3. The Response was prepared by a team of experienced engineers and other professionals from North Atlantic and from other organizations under the direction of North Atlantic management. The Response is based on the results of a series of reviews and assessments that are documented in an Engineering Self-Assessment Report (ESAR). The reviews and assessments that comprise the ESAR are described in the Response. The factual content of the Response was reviewed by a Validation Team. Management oversight was provided by a Senior Review Team consisting of senior North Atlantic managers and directors. The Nuclear Safety Audit Review Committee and the Station Operating Review Committee reviewed the Response.
4. I have read the attached Response and in reliance on that review, my inquiries of the individuals involved in the preparation of the Response, and the processes and reviews discussed in the preceding paragraph, and independent oversight, do hereby state that the contents of the accompanying Response are true and correct to the best of my knowledge, information and belief.

Ted C. Feigenbaum

TED C. FEIGENBAUM
Executive Vice President and Chief Nuclear Officer
North Atlantic Energy Service Corporation

Subscribed and Sworn
before me on this 7th day
of February, 1997.

My commission expires December 22, 1998

Susan J. Messer

Susan J. Messer
Notary Public

EXECUTIVE SUMMARY

The NRC's letter of October 9, 1996 contained six requests for information designed to elicit additional confidence and assurance that Seabrook Station is operated and maintained within its design bases and that deviations are reconciled in a timely manner. Seabrook Station's historical performance, combined with the findings of past audits and assessments and the effectiveness of current programs, processes and procedures, may have provided a sufficient basis to respond to the NRC's requests. North Atlantic, however, made a substantial investment of resources to conduct a comprehensive self-assessment to provide additional objective evidence to support its response. In addition to a review of historical performance and past audit, assessment and inspection findings, the self-assessment included the following effectiveness and verification reviews and assessments:

- An assessment of the adequacy of the linkages within important current programs, processes and procedures that deal with configuration management.

- An in-depth vertical slice review of seven risk significant systems.

- A review of nine important engineering technical programs and topical areas.

- An independent assessment of Chapter 15 (Accident Analyses) of the Updated Final Safety Analysis Report (UFSAR).

- A review to identify source documents for Technical Specification numerical values and the appropriateness of Technical Specification Limiting Conditions for Operation, Safety Limits, and Limiting Safety System Settings.

The self-assessment was conducted over a period of about two and one-half months involving a staff of about 60 professionals. More than 15,000 person-hours were expended in the effort—

about 6,500 person-hours in the vertical slice reviews alone. The engineers and other professionals performing the reviews and assessments were highly qualified and experienced. Many of our most senior and knowledgeable design engineers were assigned to the project. The average industry experience of the 30 member vertical slice review team was over 20 years.

Details on the scope, methodology, findings, recommendations and overall conclusions of these reviews and assessments have been consolidated into a North Atlantic Engineering Self-Assessment Report (ESAR). The responses to the NRC's six requests provided in this report are based on the ESAR.

The Responses to Requests (a) and (d) are primarily descriptive of current programs. The Response to Request (a) describes configuration management at Seabrook Station today including the 10 CFR 50.59 evaluation process, the UFSAR update process and the implementation of the quality requirements of Appendix B to 10 CFR 50 in the design control process. In the Response to Request (d), the current corrective action, operating experience and employee concerns programs are discussed. The response also discusses the improvements that are being made to the corrective action program and summarizes the audits and assessments that have benchmarked its evolution.

The responses to Requests (b) and (c) provide the rationales for drawing the conclusions that design bases requirements are translated into procedures and that the physical plant is consistent with the design bases, respectively. They draw heavily on the historical review that was assembled for the ESAR. The conclusion that we reach in each of these responses is that Seabrook Station's design bases rest on a solid foundation, that change has been effectively controlled, and that internal and external oversight has been diligent in identifying shortcomings in order to improve configuration management. The solid foundation was the result of the stringent acceptance criteria that had to be met by a prospective licensee by the time Seabrook Station's operating license application was submitted. An equal contributor was the systematic and thorough manner in which the utility assumed full design and configuration management

responsibility from the architect-engineer in 1987. These programs have since been managed and implemented by North Atlantic staff with relatively little work assigned to contractors.

The Response to Request (e) summarizes the basis for North Atlantic's overall conclusion on the effectiveness of the configuration management program and provides the results of the effectiveness and verification reviews and assessments noted above.

The Response to Request (f) describes the mature Design Basis Document and Engineering Design Standards programs that have been established at Seabrook Station.

North Atlantic's overall conclusion is that the results of the self-assessment indicate that there is reasonable assurance that Seabrook Station has been, and continues to be, operated and maintained within its design bases and that deviations are reconciled in a timely manner. As expected of an assessment of this scope and intensity, especially one including vertical slice reviews, deviations, discrepancies and areas requiring improvement were identified. All have been documented and will be evaluated and dispositioned in accordance with the Station's existing corrective action and action item tracking programs.

Three areas that have been selected for increased attention are UFSAR consistency, Calculation consistency, and Engineering Topical Area structure and visibility.

We had already approved and scheduled the implementation of a number of specific steps to enhance the process by which facility and procedure changes are reviewed for their impact on the UFSAR. As a result of the self-assessment, we shall conduct a common cause analysis of UFSAR discrepancies to determine what further steps may be necessary. The scope of a comprehensive UFSAR review shall be documented in separate correspondence.

A common cause analysis shall also be conducted to address the Calculation findings. It will identify common conditions and any generic issues that may exist. An action plan to improve performance in this area will then be formulated.

The results of the Engineering Technical Program and Topical Area reviews indicated that the programs, such as Environmental Qualification and Fire Protection, were more consistently implemented than the topical areas, such as Station Blackout or High Energy Line Break. As a result of the knowledge gained from the self-assessment, North Atlantic will develop a plan to provide more organizational understanding and visibility to the topical areas.

All of the Adverse Condition Reports (ACRs) generated by the self-assessment are being evaluated and dispositioned in accordance with the corrective action program requirements. Findings that did not meet the threshold criteria for an ACR and recommendations for improvement will also be formally tracked, evaluated and dispositioned. In addition, North Atlantic has contracted with Performance Improvement International (formerly FPI Inc.) to conduct a collective evaluation of the ACRs generated by the self-assessment. This will provide further assurance that we have gained all the lessons available from the results of the self-assessment, that all opportunities for program improvement have been identified, and that appropriate corrective actions have been applied to all areas of weakness.

Section VI of the report provides a consolidated list of the actions that North Atlantic is taking to improve our configuration management and corrective action programs.

TABLE OF CONTENTS

TABLE OF CONTENTS

(continued)

FIGURES

I. INTRODUCTION

This is North Atlantic's response to the NRC's letter of October 9, 1996 which requested information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of Seabrook Station's design bases information. The NRC asked for information that would provide them with added confidence and assurance that Seabrook Station is being operated and maintained within the design bases and that any deviations are reconciled in a timely manner. The NRC has issued a similar letter to other operating U.S. nuclear power plants. The requested information will be used by the NRC to verify compliance with the terms and conditions of the operating license and NRC regulations and to verify that the Updated Final Safety Analysis Report (UFSAR) properly describes the facility. The NRC's specific requests are provided below [Note: additional information requested by the NRC is designated as item (f) in this report]:

(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;

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

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

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

(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.

(f) 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 or reconstitution programs have been completed or are being conducted, provide a description of the review programs, including identification of the systems, structures, and components (SSCs), 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 SSCs and plant-level design attribute reviews, the expected completion date, and method of SSC prioritization used for the review.

The NRC required a response to the letter within 120 days of its receipt which equates to February 7, 1997 for Seabrook Station.

After a brief discussion of Seabrook Station's licensing and operating history in Section II (BACKGROUND), the methodology used to develop the responses to the NRC's requests is presented in Section III (METHODOLOGY). As described more fully therein, a three-tiered approach was taken: (1) the history of Seabrook Station, including audits and assessments, as it relates to the design bases and configuration control was reviewed and summarized; (2) the current set of programs, processes and procedures that constitute configuration control was mapped, described and evaluated; and (3) the effectiveness of these programs in maintaining the plant configuration within the design bases was measured by a set of assessments including detailed vertical slice reviews of seven plant systems. The information, findings and conclusions derived from these activities was then used as the primary source for responding to the six NRC requests. Section IV (ORGANIZATION AND REVIEW) describes the project team organization and the review and approval process. Section V (RESPONSES TO THE NRC REQUESTS) provides the responses to the six requests. Section VI (ONGOING AND FUTURE ACTIONS) delineates the efforts related to configuration management and corrective program issues that had been started prior to beginning the self-assessment. It also describes the additional actions that North Atlantic is taking as a result of the self-assessment.

North Atlantic's existing procedures and guidelines governed the activities involved in collecting, developing and analyzing the information that supports this response. The results of

reviews and assessments conducted have been consolidated into a North Atlantic Engineering Self-Assessment Report (ESAR). The ESAR, in turn, served as the primary source for responding to the NRC's six requests. This Engineering Report is available on-site for review.

II. BACKGROUND

Seabrook Station is an 1150 Megawatt electrical generation station employing a pressurized water reactor located in Seabrook, New Hampshire. Seabrook Station has eleven joint owners. North Atlantic Energy Service Corporation (North Atlantic), operates Seabrook Station on behalf of all of the joint owners. The Nuclear Steam Supply System was provided by Westinghouse; the turbine-generator by General Electric. Architect-engineering and construction services were provided by United Engineers & Constructors (now Raytheon Engineers and Constructors). The Yankee Nuclear Service Division of the Yankee Atomic Electric Company is responsible for the limiting non-LOCA accident analyses. Westinghouse performed the original accident analyses and is still responsible for the LOCA analysis.

The NRC issued a Construction Permit to Public Service Company of New Hampshire (PSNH) on July 7, 1976 for two identical pressurized water reactors at Seabrook, New Hampshire. In October, 1981, the NRC accepted PSNH's application, including a combined Final Safety Analysis Report, for an operating license for both units.

In 1984, PSNH and the other Seabrook joint owners authorized the formation of a separate division of PSNH, known as New Hampshire Yankee (NHY), to act as managing agent for the project on their behalf. When NHY took over in June, 1984, the organizational structure and reporting relationships changed but the Seabrook Station staff was essentially unaffected.

Construction of Unit 2 was effectively terminated in 1984 and its construction permit allowed to expire in October 1988.

By July, 1986, construction of Unit 1 had been substantially completed and, in October of that year, the NRC issued the initial operating license. This license, however, only authorized fuel loading and pre-criticality testing. It did not allow NHY to start the reactor or conduct pre-operational tests at low power. These limitations were imposed because of the ongoing

emergency planning litigation before the NRC licensing boards. The low power license was received in May, 1989 and the full-power license in March, 1990.

In May, 1992, the NRC approved licensing amendment requests filed by NHY seeking NRC approval of modifications to the license reflecting the assumption of PSNH's share of Seabrook Station by North Atlantic Energy Corporation and the designation of the North Atlantic Energy Service Corporation as the new managing agent. The staff and organizational structure of NHY was transferred in its entirety to North Atlantic.

III. METHODOLOGY

A. Definitions

Since the terms 'design bases', 'engineering design bases', 'licensing bases' and 'configuration management' are used throughout the response, it is important that their meanings be clear. The 'design bases' of a nuclear power plant is prescribed by regulation in 10 CFR 50.2:

> Information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system or component must meet its functional goal.

The design bases, as so defined, are contained in the UFSAR and are implemented through numerous other documents such as technical specifications, operating procedures, maintenance procedures, work control procedures and design change reports. The design bases of Seabrook are a subset of a broader category of design constraints known as 'engineering design bases.' The engineering design bases include drawings, specifications, calculations and requirements of other regulatory bodies (EPA, the State of New Hampshire etc.), industry codes and standards, good engineering practices, and other considerations that North Atlantic is required, or has chosen, to apply to optimize system design for safety, efficiency or reliability.

The design bases are also a subset of the licensing bases. The licensing bases consist of the total set of information upon which the NRC based its comprehensive determination that the design, construction and operation of Seabrook Station met the standards and requirements for issuance of an initial operating license. It also includes commitments made to the NRC in response to continuing regulatory activities or as a result of North Atlantic's own initiatives and upon which the NRC bases its continuing approval of the operating license. Included in the licensing basis,

for example, would be responses to NRC generic letters and bulletins, and commitments made in response to NRC inspections and in responses to Notices of Violations.

As illustrated in Figure 1, the design bases is effectively the intersection of the engineering design bases and the licensing bases.

The terms 'configuration management' or 'configuration control program', as used in this response, refer to the set of programs, processes and procedures that collectively provide the controls necessary to ensure that the physical and functional attributes of Seabrook Station are consistent with the established design configuration; that the plant is designed, constructed, maintained and operated consistent with the design bases and that the information describing these attributes is readily accessible and maintain the plant within its engineering design bases. Design control, the 10 CFR 50.59 safety evaluation process, the UFSAR update program, the corrective action program and other programs and processes are all under the umbrella of configuration management, as used in this report, although corrective action is often cited separately because of the special emphasis on it in the NRC's letter and in this response. It is configuration management, for example, that dictates that if a physical change is to be made to the plant, an evaluation must be conducted to determine its impact on safety and to ensure that it is allowable under the existing license conditions and requirements. Configuration management then provides the process for communicating this change to affected operating, maintenance, testing and surveillance procedures and for ensuring that the UFSAR, and other potentially impacted documents, are revised as necessary. Configuration management also encompasses the programs necessary to ensure that the modification is properly installed and tested. This linkage is presented schematically in Figure 2 which shows the major functional areas that comprise the configuration control program.

B. Review and Assessment Methods

The NRC's letter of October 9, 1996 and its six requests focus on design bases. Of necessity, our responses are often more broadly based, frequently encompassing licensing and engineering bases. The reason for this is simply that the scope of many of the programs, processes and procedures that control plant configuration include the licensing and/or engineering design bases. Seabrook Station's design control program, for example, is set forth in the Design Control Manual (DCM). The DCM is applicable to all systems, structures and components installed inside (and including) the protected area boundary (safety or nonsafety related). It also applies to many related and interfacing systems outside the protected area.

The responses to the NRC's requests encompass the following areas:

- a review of Seabrook Station's **historical performance** with respect to configuration control;
- an assessment of **current programs, processes and procedures** and
- assessments that tested the **overall effectiveness and verification** of configuration management.

Teams were assembled to conduct the reviews and assessments in each of these areas. The responses to the six NRC requests were based on the reports of the results of their reviews. The methodologies used by the teams are summarized below. To provide perspective, the results of important past assessments on the adequacy of the design bases, and on the configuration management program that maintains it, have been identified and incorporated into the response, as appropriate.

C. Historical Performance Review

In order to provide a high level of confidence that a plant is being operated within its design bases, it is necessary to demonstrate that they were adequately established in the first place. This requires, as a minimum, a recapitulation or history of the steps taken to establish or to re-baseline the design bases. For an older plant, the starting point may be initial licensing or a later reconstitution of the design bases. Since Seabrook Station has been operating for only six and one-half years, the time of turnover of the plant from the architect-engineer and constructor to the operating organization was chosen as a logical starting point for the Historical Performance Review.

In the Historical Performance review phase, the actions taken to initially establish key elements of the design bases reviewed include:

- the establishment and control of the FSAR and the Technical Specifications,
- the translation of licensing and design bases requirements into operating, maintenance and testing procedures,
- the development of the 10 CFR 50.59 processes for proposing changes to the plant or the plant's programs from that described in the UFSAR, and
- the development of the engineering design change processes including turnover reviews, verifications and walkdowns.

To provide perspective, audits and assessments that deal with configuration controls were also reviewed. These included self-assessments, NRC inspections, INPO evaluations, other third-party inspections and routine quality assurance audits.

The results of the historical performance reviews are included in the ESAR and incorporated, as applicable, in the responses to NRC requests (b), (c) and (d) which are contained in Section V.

D. Current Programs, Processes and Procedures Assessment

The assessment of current programs, processes and procedures analyzed the logic and the linkages that exist between and among the various elements of configuration control. One of the first steps in the assessment was to develop detailed process maps for key elements of the program. Maps were developed for processes that implement 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR 50, as well as those that implement our corrective action and design change programs. These maps depict the links that exist between the programs and procedures as well as steps within procedures. An example of a link would be the procedural requirement that all design changes undergo a 10 CFR 50.59 screening evaluation. The two process maps that are shown in this submittal (Figures 5 and 6) are overviews. More detailed maps, which were used in conjunction with the assessment, are included in the ESAR.

The link may consist of a program, a procedure or steps in a procedure. The project team conducted a systematic review of each link and the programs, processes and procedures that each connects, against the following set of attributes which are indicative of an effective program:

- The program, or procedure has a clearly defined owner (department, individual, etc.) and the responsibility for the implementation steps have been clearly defined and assigned.
- The program, or procedure has a clear beginning and end with well defined interfaces, timing requirements, and internal communications links that support an easy-to-follow flow path.
- The program or procedure product or output data is consistent with the front end objective and receives the appropriate level of review including a screening or evaluation in accordance with 10 CFR 50.59 when appropriate.
- The program or procedure does not circumvent other programs, processes, or procedures.
- The program or procedure is being used.

The assessment included 'table top' exercises in which hypothetical changes were entered into each of the processes. By following the steps dictated by the written procedural or programmatic requirement depicted in the detailed process maps, the evaluators were able to determine if the processes themselves, without reliance on the resourcefulness of the user, provide adequate assurance that the engineering design bases are maintained as changes are made. In this respect, the assessment complements the other verification activities such as the vertical slice reviews. While their sampling methods provide symptomatic indication of the effectiveness of configuration management, this assessment measures procedural adequacy. In other words, it helps to establish whether the current state of the design bases exists because of the configuration management program or in spite of it. The results of this assessment are discussed in Section V in the Response to Request (e).

E. Overall Effectiveness and Verification Assessments

In the Overall Effectiveness and Verification phase, the effectiveness of the configuration control program in ensuring that the plant has, in fact, been maintained and operated within its design bases has been tested. The tests consisted of **vertical slice reviews** of selected safety significant systems, an **engineering technical programs assessment** of nine engineering programs and topical areas against their design and licensing bases, an **UFSAR Chapter 15 (Accident Analyses) review**, and a **Technical Specification Review**. These are described below.

1. Vertical Slice Reviews

The vertical slice review methodology is based on the NRC's Safety System Functional Inspection (SSFI) and uses an in-depth, multi-disciplined engineering approach to assess plant systems. Each system is evaluated for the accuracy and completeness of design and licensing bases documents, for the conformance of design with licensing and design requirements, for the proper translation of the design bases into operating procedures, for the effectiveness of plant procedures (operating, maintenance and testing) in maintaining the system within its design bases, and for the conformance of the as-built system to the design requirements.

The seven systems subjected to a vertical slice review are all dominant contributors to a set of accident sequences that account for a significant portion of the reactor core damage risk according to the Seabrook-specific Probabilistic Safety Assessment Study. The systems selected were:

- 125 Volt DC System
- Primary Component Cooling Water System (PCCW)
- Emergency AC System
- Emergency Diesel Generator System (DG)
- Emergency Feedwater System (EFW)
- Plant Protection System
- Residual Heat Removal System (RHR)

The vertical slice reviews covered the set of 27 topical areas shown in Table 1. Based on the functional areas delineated in the NRC's Safety System Functional Inspection Manual, they collectively form a template that defines the engineering design bases.

A set of attributes for each individual vertical slice system was developed and evaluated against these topical areas. Figure 3 is an example of the matrix form used to keep track of the system attributes selected for a vertical slice review and the topical areas against which they were assessed.

2. Engineering Programs and Topical Areas Assessments

The purpose of this assessment was to provide an indication of the overall effectiveness of our configuration management program in ensuring that the engineering programs and topical areas continue to meet licensing requirements and commitments and reflect the engineering design bases. The nine Engineering Technical Programs and Topical Areas reviewed were selected

because individually and collectively they have the potential for significant impact on the engineering design bases and encompass a broad spectrum of structures, systems and components. The programs and topical areas selected are:

Programs

- Environmental Qualification Program
- Fire Protection Program (10 CFR 50 Appendix R)
- Testing Programs (Inservice and 10 CFR 50 Appendix J)
- Inservice Inspection Program (ASME Section XI)

Topical Areas

- Electrical Separation
- Post Accident Monitoring (Regulatory Guide 1.97)
- Individual Plant Examination of External Events (IPEEE)
- High Energy Line Break (HELB)
- Station Blackout (SBO)

The team reviewed the licensing and design requirements for each program and topical area. This typically was found in the UFSAR, plant Technical Specifications, the NRC Safety Evaluation Reports or in correspondence containing licensing commitments. The implementation of these requirements was then sampled for technical adequacy. Finally, the programmatic controls for maintaining the plant within the design bases dictated by these requirements and commitments were evaluated.

3. UFSAR Chapter 15 Assessment

Chapter 15 of the UFSAR was chosen for this assessment because it contains the accident analyses that define the boundaries for the configuration and operation of Seabrook Station. The plant must remain within the bounds set by the input assumptions of these analyses. To a large

degree, therefore, the Chapter 15 analyses govern the configuration of the plant through output documents such as Technical Specifications and are essential to the establishment of the design bases and a baseline for the configuration management program.

The assessment was performed by an independent contractor (Scientech, Inc.) that does not have responsibility for the calculations of record for Seabrook Station. It was supported by Yankee Atomic Engineering Company, Nuclear Services Division, the organization responsible for the non-LOCA and radiological analyses and by Westinghouse Electric Company which is responsible for the LOCA analysis.

Scientech reviewed the UFSAR Chapter 15 to determine if it is consistent in format and content with Regulatory source documents such as Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports), NUREG-0800 (Standard Review Plan for the Review of SARs), and the NRC's Safety Evaluation Reports on Seabrook Station. An integrated assessment of UFSAR Chapter 15 supporting documentation was performed to assure that the input documents are consistent with the information appearing in the UFSAR, and to verify that a sampling of the output documents are consistent with the assumptions and limits established by the UFSAR analyses. From this review, Scientech developed a set of 'Potential Impacts' that the source documents have on Chapter 15. Potential impacts are licensing or design basis inputs and requirements that should be reflected in the UFSAR. They include such things as key assumptions, system and component parameters and conditions, protection equipment setpoints and delays, and operator actions. Scientech then conducted an integrated review of the collected information to determine if the UFSAR Chapter 15 accident analyses satisfied the requirements of the source documents and if a sampling of the output documents were, in turn, consistent with the UFSAR.

4. <u>Technical Specifications Review</u>

The Technical Specifications set the bounds for operating the plant. They specify the minimum plant configuration required for different modes of operation, from cold shutdown to full power,

in terms of equipment and system operability, instruments and staff. The review conducted for this response was in two parts. One part consisted of a review of the Safety Limits, Limiting Safety System Settings and Limiting Conditions for Operation (LCOs) while the other reviewed the bases for a large sample of the numerical values found in the Technical Specifications.

The LCOs were checked to ensure that they accurately described the system or equipment for which they were setting a limiting condition or minimum required configuration and that the system design was consistent with the LCO. The team also reviewed the LCOs for consistency with the UFSAR Chapter 15 accident analyses upon which they are based. Although the Technical Specification action statements and surveillances were not within the scope of this review, their accuracy and consistency were tested through the vertical slice reviews.

In the Technical Specification numerical values (flows, voltages, volumes, pressures, etc.) review, values that did not have an adequately identified source document or basis were determined. An effort was then made to obtain the source document or basis. Numerical values which were derived from Standard Technical Specifications were also reviewed for appropriateness. The information and documentation collected from this effort will be incorporated into the ongoing effort to convert the Technical Specifications to the Improved Standard Technical Specification format.

IV. ORGANIZATION AND REVIEW

The self-assessments were completed by three implementing teams with management oversight from a Senior Review Team (SRT). The three implementing teams were designated the Historical Performance Team, the Current Programs, Processes and Procedures Team, and the Effectiveness and Verification Team. The Project Manager was the North Atlantic Director of Engineering. Each of the Team Leaders were individuals with from 15 to 20 years of experience in the nuclear power industry. The basic functional organization for the project is show in Figure 4.

The SRT provided management direction on the activities being performed to respond to the 10 CFR 50.54(f) letter. It was chaired by the Station Director and consisted of the Station's senior managers and two outside consultants to provide an independent view. The SRT's scope included:

- the assessment methods used,
- the criteria used in the selection of the systems, components, procedures, and programs to be assessed,
- the individual Assessment Plans and assessment results, and
- the response to the NRC.

Quality Assurance provided in-process oversight which included review of the qualifications and training of the project team. In addition, a validation team reviewed the ESAR for accuracy and consistency and reviewed this response for accuracy and the supportability of its conclusions.

This response to the NRC was also reviewed by the Nuclear Safety Audit Review Committee and reviewed and approved by the Station Operations Review Committee.

V. RESPONSES TO THE NRC's REQUESTS

A. 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 5C.*

Seabrook Station's engineering and design control processes, including those that implement 10 CFR 50.59 and 10 CFR 50.71(e) meet the applicable quality assurance requirements of Appendix B to 10 CFR 50. The response to this request will provide an overview of these processes focusing especially on design control, the 10 CFR 50.59 evaluation process, and the UFSAR update process (10 CFR 50.71(e)).

The Station Manuals system, described in UFSAR Section 17.2, provides the programs, processes and procedures that are integral to configuration management. Some of the key manuals are, therefore, identified and discussed in terms of their roles in the design and configuration control processes. The Design Control Manual (DCM) is explored in greater detail in order to provide an overview of the Engineering design activities that effect change in the Station.

The 10 CFR 50.59 and 10 CFR 50.71(e) processes are important elements of the design control process. Their intersections with that process are reported in the discussion of the DCM. They also play a wider role in that they safeguard the design bases during Station changes other than those generated under the auspices of the DCM, such as procedural changes. A separate generic description of these processes at Seabrook Station is, therefore, provided.

Relevant interfaces of the 10 CFR 50 Appendix B quality assurance program, as addressed in UFSAR Section 17.2 and the Operational Quality Assurance Program, are discussed in parallel with the Station Manuals System and the Design Control Process.

1. The Seabrook Station Manuals System

The Station Manuals system establishes the organizational policies, goals, responsibilities and administrative programs necessary to maintain the engineering design and configuration control process. The hierarchy of the Manuals System goes from Management Manuals to Program Manuals to Reference Manuals. Management Manuals address Seabrook Station organizational structure, policies and goals. Program Manuals provide the administrative processes and controls for functions which require interaction between groups and departments. Reference Manuals provide supporting information to carry out the administrative functions.

a. Management Manuals

The following three Management Manuals dictate Seabrook Station's overall operational program direction:

- North Atlantic Management Manual (NAMM)
- Station Management Manual (SSMM)
- Operations Management Manual (OPMM)

The NAMM provides the ownership description of Seabrook Station, sets forth organizational responsibilities and lists of recognized committees. It captures management policy statements and summarizes the plans and programs that exist to implement the Station's policies and goals, including our license commitments, in a consistent manner. The NAMM also includes administrative procedures that are applicable to the entire North Atlantic organization but do not reside in a program manual. One such procedure identifies the requirements for preparation, review, approval, revision and cancellation of manuals and procedures within the manuals system. Thus, changes to programs, such as the Design Control Manual, the 50.59 evaluation process or UFSAR maintenance, are administered to consistent expectations.

The Station Management Manual (SSMM) addresses the overall Station approach to activities affecting the configuration of structures, systems and components at Seabrook Station. A primary focus is the preparation, approval and revision of administrative procedures, operating procedures and preventative maintenance documents that maintain implementation consistency. The Station Operation Review Committee (SORC) functions and responsibilities, as required by Technical Specifications, are identified in the SSMM. SORC reviews all safety evaluations conducted under the provisions of 10 CFR 50.59. The SSMM references interfacing Program Manuals that provide links to Planning and Scheduling, Maintenance, Operating Experience and Technical Training Programs.

The Operations Management Manual (OPMM) addresses the Operations Department organizational structure, responsibilities, shift compositions, shift operations, and procedure compliance expectations. Also included are the administrative controls on abnormal and emergency operating procedures, configuration controls and mode changes. A primary focus is maintaining cognizance of the Station's configuration and operating the Station within the requirements of the license.

Operations department concurrence is obtained for design and licensing bases changes affecting Station operation. Operations Department procedure revisions are generally needed to implement significant changes. These revisions must be approved before a design change is declared operable. The Operations Department Procedure Coordinator determines if design changes have a procedural impact. The Operations Technical Projects Group reviews design changes for general operational impacts, including human factors considerations and impact on log sheets.

b. Program Manuals

Program Manuals address program implementation responsibilities and tasks necessary to carry out the requirements of the Management Manuals. Many directly address the engineering, design and configuration control processes. Four Program Manuals (Regulatory Compliance,

Operating Experience, Procedure Administration and Quality Assurance) are summarized below. The Design Control Manual is discussed in more detail in Subsection 2.

Regulatory Compliance (NARC)

The NARC includes responsibilities and program scope for regulatory compliance. The key processes discussed include maintenance and update of the UFSAR, making changes to the Station under 10 CFR 50.59 provisions, identification of new commitments and changes to existing commitments. Guidance is also available for reportability determination of Station activities or configurations identified by the ACR process. The 10 CFR 50.59 guidance is universal for all Station users.

Operating Experience Manual (SSOE)

The SSOE defines the responsibilities and actions to document and resolve Adverse Conditions and address internal and external operating experience (see also Response to request (d)). Adverse Conditions are documented in an Adverse Condition Report (ACR). The ACR process is the primary means to document configuration discrepancies. Each issue is judged against threshold criteria and routed through a planned sequence in the organization. Specific groups determine the validity of an issue, impact on Operability of systems, structures and components, reportability to regulatory authorities and determination of corrective action assignments.

The ACR process is regularly reviewed for trends, program effectiveness and adequacy to prevent problem recurrence. This program fulfills the corrective action requirements of 10 CFR 50 Appendix B and the Operational Quality Assurance Manual (NAQA).

Procedure Administration Manual (NAPA)

The NAPA describes responsibilities and provides guidance for writing procedures. An important element of this guidance is a listing of approved "Action Words" with precise

definitions. This ensures that the procedural expectation is clear in order to minimize personnel error. As part of the review process defined by NAPA, a Subject Matter Expert (SME) is assigned responsibility for assessing UFSAR and technical specification impact as well as research of design changes and other source documents. A 10 CFR 50.59 screening evaluation is required for new or revised procedures.

Quality Assurance Manual (NAQA)

The NAQA defines the management controls for materials and activities within the scope of 10 CFR 50, Appendix B and UFSAR Section 17.2. It comprises those planned and systematic actions necessary to provide adequate confidence that structures, systems and components will perform satisfactorily in service. Key elements of the NAQA address the Engineering design and configuration control processes. These elements include Design Control, Procurement Control, Manuals, Procedures, Instructions and Drawings, Corrective Action and Nuclear Records. Additionally, the NAQA provides reference to applicable regulatory guides addressing these program elements.

c. Protected Steps In The Manuals Program

The administrative guidance for the manuals system provided in the NAMM requires that specific text or steps that are added to manual chapters or procedures as a result of regulatory or operating experience documents and commitments shall be protected. This requirement is designed to ensure that the text or steps are not altered or inadvertently removed without proper justification or documentation. Removal, cancellation or modification of text that is protected as a result of a regulatory commitment can only be done under the Commitment Change Process, which includes a 10 CFR 50.59 evaluation.

2. The Design Control Manual and the Design Control Process

Section 17.2 of the Seabrook Station UFSAR (Quality Assurance During the Operations Phase) contains North Atlantic's policies on quality and commitments to the applicable regulatory requirements of 10 CFR 50 Appendix B including those related to design control. The NAQA defines the Operational Quality Assurance Program (OQAP) and the associated management controls for materials and activities designed to implement these policies and commitments. The provisions related to design control are incorporated into the responsibilities and action steps in the Design Control Manual (DCM).

Design control processes and responsibilities are set forth in the Design Control Manual (DCM). The requirements of the DCM are applicable to structures, systems and components required to comply with 10 CFR 50 Appendix A (General Design Criteria), 10 CFR 50.34(b), 10 CFR 50.47 and licensing commitments in the Updated Final Safety Analysis Report. The same design control process is used for design work beyond the scope of these regulatory requirements including all systems, structures and components installed inside (and including) the protected area boundary, safety or non-safety related, as well as many related and interfacing systems outside the protected areas.

a. Responsibilities

The Director of Engineering has overall responsibility for the design control program including its scope and content, approval of new and revised designs and maintenance of design bases documentation to support operation.

The Station Director approves implementation of design changes that require a 10 CFR 50.59 safety evaluation. Responsibilities include the review, scheduling and processing of design changes. The physical configuration control of the structures, system and components also rests with the Station Director.

The Oversight organization is responsible for audit, surveillance and inspection of activities supporting the design control process.

SORC and NSARC review design changes within the scope of their respective charters.

b. Design Change Types

Design Change Records (DCRs) and Minor Modifications (MMODs) are the two principal types of design change documents. DCRs are typically large and/or complex modifications, safety or non-nuclear safety related, that have the potential for affecting safety related structures, systems or components. MMODs are typically non-complex/minor changes affecting either safety or non-safety related structures, systems or components. Both DCRs and MMODs must undergo 10 CFR 50.59 screening. If the MMOD screening indicates that a safety evaluation is required, the proposed change must be converted to a DCR which carries more rigorous review and documentation requirements. The safety evaluation is part of the DCR and is reviewed and approved with it.

DCRs and MMODs require a cover sheet that includes the 10 CFR 50.59 screening evaluation, an executive summary and the approval signatures. The narrative section includes the bases of the current design, the method of change, design requirements and references. The design input section addresses the major technical documents such as UFSAR changes, license commitments, design bases documents as well as codes, standards, regulatory requirements and design details that describe the existing structure, system or component.

A detailed design section describes the changes in sufficient detail to demonstrate that they satisfy the design inputs. Implementation and operational consideration sections are added to highlight specific design issues that are critical to the success of the design. For example, mode applicability restrictions may have to be in place in order to implement the design change or operational restrictions may need to be included in procedures before the change can be declared operable.

Certain Station enhancements, including configuration changes which do not meet the threshold of an MMOD or DCR, are effected through a Maintenance Support Engineering Evaluation (MSEE). MSEEs are used when the plant requires a minor change that has no safety significance. A kick plate added to a platform is an example of an MSEE. The MSEE form requires the originating engineer to document the applicability of the 10 CFR 50.59 process. If a safety evaluation is required, an MSEE cannot be used to effect the change. MSEEs are also screened by the Design Engineering Manager to ensure it is the appropriate vehicle for the change. They are implemented through work control procedures.

Station personnel frequently have requests for design work, have questions on design intent or just need clarification on engineering and design issues. The vehicle used to communicate with engineering in these situations is the Engineering Work Request (EWR). Engineering scopes new design work based on the EWR description and any other operational information gathered during the evaluation conducted in response to the EWR. EWRs are screened to determine if problems or issues presented in them meet the threshold criteria for documentation in the corrective action program. Since plant changes are not made with an EWR, a 10 CFR 50.59 screening is not required.

c. <u>The Design Change Process</u>

Once the decision is made to go forward with a design change, the design change process proceeds in the following sequence:

- Design change development
- Design change review
- Design change approval
- Design change implementation
- Design change closeout

DESIGN CHANGE DEVELOPMENT—Each design change is assigned to a cognizant North Atlantic engineer who coordinates the development approval, and revisions to a design package. Design activity begins with confirmation of the approved scope and Engineering schedule. The interaction of the engineering disciplines and required interfaces with other internal and external organizations as well as any restraints on the design deliverable are included on the schedule.

In this phase, engineers from the appropriate discipline develop new design details or mark-up specific changes to existing design bases information. Depending on the scope of the design change, new or revised calculations, specifications, and/or drawings may need to be created or revised. The DCM addresses the administrative controls necessary to control these changes. The steps taken during this phase conform to the requirements of Appendix B to 10 CFR 50 to establish measures to identify and control design interfaces and coordinate participating design organizations.

In accordance with the DCM, the cognizant engineer researches the change to determine if the Operating License, Technical Specifications, Technical Requirements, the UFSAR or any design document is affected. The findings must be documented in the DCR or MMOD. This results in a detailed listing of affected documents. When changes are needed in Licensing documents, the applicable procedure for a License Amendment, Technical Specification change, UFSAR change, etc. govern the change approval process. UFSAR changes are included in the design change package and considered a part of the design change package. In this way, the proposed design change itself and the UFSAR change that it generates can be reviewed concurrently. This encourages a more thorough and timely evaluation of UFSAR impact than would be the case if the UFSAR change were submitted separately after approval of the design change.

The change package submitted for review and approval also contains the results of the 10 CFR 50.59 applicability screening. The screening questions determine if the change affects the UFSAR or the Operating License (see Section 3 below). Each question requires a written basis for the determination.

DESIGN CHANGE REVIEW—The DCM sets the requirements, standards and guidelines for design verification. Design verification may be accomplished by design reviews, alternate calculation or qualification testing. These interdiscipline, interdepartmental, and independent reviews provide reasonable assurance that the design change package is either within the existing design bases or revises the design bases in accordance with the design control and 10 CFR 50.59 evaluation process prior to submittal for approval.

The design engineers submit the prepared DCR or MMOD package for review. The package contains the 10 CFR 50.59 screening and/or safety evaluation and an UFSAR change request if needed. The reviewers are designated by the DCM and include the design engineering disciplines, the system engineer, an independent reviewer and external reviewers, such as the Operations and Maintenance Departments.

In this phase, the DCM addresses the requirements in Appendix B to 10 CFR 50 to provide measures for verifying and checking adequacy of design.

DESIGN CHANGE APPROVAL—The design engineer resolves all questions and comments from the review process and updates any information, as required. Approval of design changes depends on the type of design change package. Both an MMOD and a DCR receive System Engineer, Independent Reviewer and Engineering management review and approval. A DCR also requires SORC review and approval. Upon approval, the DCR or MMOD is distributed via controlled distribution to the appropriate staff responsible for implementation of the design. These provisions meet the 10 CFR 50, Appendix B requirement for control of design change information affecting quality.

DESIGN CHANGE IMPLEMENTATION—Design changes are implemented under the Work Control Program. Work control documents identify the DCR or MMOD as the initiating document and provide a detailed sequence of activities necessary to properly implement the

design change. The Implementation Engineer confirms that design features have been satisfactorily tested and completed in the work control documents.

Approved DCRs and MMODs may be changed via the DCN process. DCNs are categorized as either Engineering, Field or Administrative. Engineering DCNs alter the intent, QA requirements or the safety evaluation of the design. A screening is conducted on DCNs to determine if a 10 CFR 50.59 safety evaluation is required. A Field DCN allows work to proceed after the design engineer approves the change or engineering solution and the engineering records are appropriately documented. A Field DCN must not extend the design change beyond its previously evaluated scope. The 10 CFR 50.59 screening or safety evaluation on the original DCR is reviewed against the scope of the Field DCN to confirm that advance approval is not required. An Administrative DCN is used to correct minor administrative issues after the design change is closed. It requires approval from the Design Engineering Manager to ensure that the original 10 CFR 50.59 screening and or safety evaluation is not invalidated, and that the change requested is, in fact, an administrative issue. The DCN process ensures conformance with the Appendix B to 10 CFR 50 criterion that design changes, including field changes, shall be subject to design control measures commensurate with that of the original design.

DESIGN CHANGE CLOSEOUT—Upon completion of all implementing work control documents, a design change will be declared Operable by the Operations Shift Manager. The operability process ensures that the design implementation is complete and the new or modified structures, systems and components are capable of performing their design function. As part of the operability process, Engineering revises and distributes Category 1 documents to reflect the completion of the design installation. Category 1 documents are those that are essential to, and provide direct support of, Station operations.

Upon declaration of Operability, the Implementation Engineer ensures that affected department procedures and activities have been updated or tracked as an action item in the Station's commitment tracking system. The full Operability declaration also triggers the distribution of any associated UFSAR change to the Approved UFSAR Change Reference (AUCR) manuals.

These manuals are distributed throughout the Station organization for interim use until the periodic UFSAR update submittal is made to the NRC.

3. The 10 CFR 50.59 Safety Evaluation Process

Procedural requirements and guidelines for performing 10 CFR 50.59 evaluations are found in the Regulatory Compliance Manual (NARC). The NARC includes guidelines as set forth in NSAC-125 (Guidelines for 10 CFR 50.59 Reviews and Evaluations, June 1989). The North Atlantic Management Manual (NAMM), Station Management Manual (SSMM) and the Procedure Administration Manual (NAPA) identify those procedures that require a 10 CFR 50.59 screening and safety evaluation.

A simplified schematic of the 10 CFR 50.59 process, as applied to design changes, is provided as Figure 5.

The 10 CFR 50.59 Evaluation is the process of determining whether or not a proposed change or activity constitutes an unreviewed safety question or requires an amendment to the Operating License. There are two parts to the evaluation: the determination of safety evaluation applicability, or screening, and the safety evaluation itself.

The screening part of the evaluation consists of the following questions:

- Does the proposed change make changes in the facility as described in the UFSAR?
- Does the proposed change make changes in procedures as described in the UFSAR?
- Does the proposed change involve tests or experiments not described in the UFSAR?
- Does the proposed change require a change to the Operating License or are additional Operating License requirements needed?

In addition to documented responses on the form, Seabrook Station's process requires a written basis for the answers to the screening questions for design changes. A forthcoming revision to the NARC will also require a written basis for new procedures and revisions.

If the answer to any of these questions is 'Yes', a safety evaluation must be conducted. The Safety Evaluation requires a written response to the following seven questions:

- Will the probability of an accident previously evaluated in the UFSAR be increased?
- Will the consequences of an accident previously evaluated in the UFSAR be increased?
- Will the probability of a malfunction of equipment important to safety be increased?
- Will the consequences of a malfunction of equipment important to safety be increased?
- Will the possibility of an accident of a different type than any previously evaluated in the UFSAR be created?
- Will the possibility of a malfunction of a different type than any previously evaluated in the UFSAR be created?
- Will the margin of safety as defined in the basis for any technical specification be reduced?

If the answer to any of these questions is 'Yes', an unreviewed safety question exists and prior NRC approval of the change must be obtained.

Either the preparer or independent reviewer shall have completed the 10 CFR 50.59 Evaluation training course. The training is based on the requirements and guidance contained in the NARC which in turn is based on the NSAC-125 guidelines.

The SORC reviews 10 CFR 50.59 Safety Evaluations to verify their completeness and adequacy and to render a determination as to whether the change, test or experiment involves an unreviewed safety question. SORC provides a recommendation to the Station Director and documents the results in SORC meeting minutes. The Nuclear Safety Audit Review Committee

(NSARC) also reviews 10 CFR 50.59 safety evaluations, as required by Technical Specifications, and monitors them for quality in accordance with their charter.

In conjunction with the periodic UFSAR update submittal, North Atlantic provides a report to the NRC of changes made in accordance with the requirements of 10 CFR 50.59. This report includes a description of significant changes for which a safety evaluation was performed and which may or may not have required the UFSAR to be updated. Additional safety evaluations performed for conservatism are also listed in the report.

4. Updated Final Safety Analysis Report (10 CFR 50.71)

The Updated Final Safety Analysis Report (UFSAR) contains the design bases. It is a living document with revisions submitted to the NRC normally following completion of each refueling outage, pursuant to 10 CFR 50.71(e).

The following is a summary of the UFSAR change process. A simplified schematic of the process is provided as Figure 6.

a. Updated Final Safety Analysis Report Change Request (UFCR)

The 10 CFR 50.59 evaluation screening questions establish the relationship of the proposed change to the UFSAR. Should any one of the four questions receive a "Yes" response, a UFCR may be required. If an UFCR is required, a unique UFCR number will be assigned. The UFCR cover sheet records the affected UFSAR section, table and/or figure number. Changes that affect the Fire Protection Branch Technical Position (BTP) APCSB 9.5-1, Appendix A to BTP APCSB 9.5-1 (Appendix A) and 10 CFR 50 Appendix R Reports (Appendix R), also use the UFCR change process to capture proposed changes.

For design changes, the 10 CFR 50.59 evaluation, UFCR forms and marked up text, figures and tables accompany the review and approval process outlined in the DCM. The UFCR is reviewed by SORC and approved by the Station Director, concurrent with the design change. This approval process allows review groups to evaluate UFSAR impact and design change scope as a single action.

UFCRs not generated by a design change (stand alone) undergo a safety evaluation and require review by SORC and the approval of the Station Director.

b. Engineering Review

Engineering review of UFCR packages are performed concurrent with the design change under the scope of the DCM. Engineering reviews UFCRs to ensure conformance with design documentation and verifies that applicable commitments are properly incorporated. Stand alone UFCRs, i.e. those that are not generated by a design change package, follow a similar approval process, including Engineering review.

c. UFCR Effectiveness

UFCRs not generated by a design change are incorporated into the next quarterly revision of the Approved UFSAR Changes Reference Manual (AUCR) after approval. The AUCR is a manual that contains approved UFCRs pending their incorporation into the UFSAR as part of a periodic update. UFCRs associated with design changes become effective after the design is declared fully operable. At that time they are made available for incorporation into the AUCR Manual. The AUCR is a supplement to the UFSAR and is considered to be a part of the UFSAR in 10 CFR 50.59 evaluations for impact on the design and licensing bases of Seabrook Station.

c. Revising The UFSAR

Following submittal of each periodic UFSAR update to the NRC, the AUCR Manual is purged of those UFCRs that were incorporated into the UFSAR revision. The UFSAR update is submitted to the NRC on a page by page basis. The affected pages, figures and tables are from approved UFCRs in the AUCR and are current up to 6 months prior to the submittal date. The UFSAR update is submitted to the NRC and controlled distribution is made per the North Atlantic Records Management Manual (NARM).

Conclusion

Seabrook Station has a well-established and mature configuration management program. It includes an effective design control safety evaluation process and UFSAR update processes that meet the requirements of Appendix B to 10 CFR 50. As discussed in the Responses to Requests (b) and (c), North Atlantic assumed control of design in 1987, almost three years before full-power licensing. The process of assuming responsibility was thorough and systematic, establishing a solid baseline configuration. Since that time, the design control processes summarized in this section have served to maintain the plant within its design bases. Areas of relative weakness and deficiencies identified during previous audits and assessments as well as during the assessments conducted for this report have been, and are now, being used to improve the program.

B. Request (b): *Rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures.*

Summary

North Atlantic concludes that design basis requirements were effectively translated into the initial issue of operating, maintenance and testing procedures, and that the change control process has effectively preserved design basis requirements in procedures. While we identified instances in which specific design basis requirements were not fully translated into or preserved in procedures, none are indicative of a programmatic breakdown. Corrective actions implemented as a result of evaluations of past discrepancies have improved our performance in maintaining procedures technically correct and supportive of the design basis. Similarly, the knowledge gained from this self-assessment has already identified specific areas where improvements will be made. The collective evaluation that will be conducted on the ACRs generated by the self-assessment (see Section VI), moreover, will help to ensure that areas warranting additional attention and improvement are identified.

The following are the foundations upon which we base our conclusion that design basis requirements have been translated into and continue to be appropriately preserved in Seabrook Station's operating, maintenance and testing procedures:

- Administrative and operating procedures[1] were initially prepared, reviewed and approved by (or under the close supervision of) people who had previous working experience at or in direct support of one or more commercial nuclear power plants.
- Design basis information was obtained from the FSAR and other applicable documents and incorporated into the initial issue of procedures. The initial procedure

[1] As will be explained further, "administrative procedures" exist under different formats at Seabrook Station. For purposes of this letter, administrative procedures are defined as the collection of documents which implement, as a minimum, the administrative controls required by the Technical Specifications, Section 6. Also for purposes of this report, procedures for equipment operation, maintenance and testing intended for use during commercial operation are collectively referred to as "operating procedures."

writers, reviewers and approvers established a close working relationship with people responsible for the design, construction and initial testing of Seabrook Station. This relationship helped ensure that procedure writers, reviewers and approvers were aware of applicable design basis requirements.

- Administrative controls have historically required that operating procedures include steps intended to implement, preserve and verify the design basis of Seabrook Station. Administrative controls governing procedure preparation, review and approval provide multiple checks and balances designed to ensure that applicable design basis requirements are met.
- Operating procedures are prepared or revised by subject matter experts. Prior to approval, operating procedures may receive an inter-disciplinary review and are independently reviewed by a qualified reviewer. Procedures determined to require a safety evaluation and selected administrative procedures are also reviewed by the SORC.
- Proposed revisions and changes to operating procedures are evaluated per 10 CFR 50.59.
- A system of internal oversight programs have been implemented. Audits, surveillances, inspections assessments and reviews are performed by organizations such as Nuclear Oversight and the Nuclear Safety Audit and Review Committee (NSARC). External organizations such as INPO, the NRC and other third parties have also conducted frequent oversight activities. These oversight activities have generally concluded that our programs and procedures have been effective in maintaining the plant within its design basis.
- Our corrective action program has effectively identified procedure deficiencies, analyzed their cause and required corrective actions to prevent recurrence.

Information in support of the above foundations is presented below.

1. <u>Initial Preparation of Administrative and Operating Procedures</u>

Seabrook Station was not licensed for initial nuclear fuel loading until 1986, and was not licensed for full power operation until 1990. However, the first Station Superintendent was named in 1976. Soon after this position was filled, the Assistant Station Superintendent was named. By late 1979, additional managers, supervisors and other key members of the permanent Station Staff had been named. Most people who initially filled the key Station Staff positions had previous working experience at or in direct support of one or more commercial nuclear power plants. In summary, the initial Seabrook Station Staff, as a group, possessed the requisite knowledge and experience to understand the importance of ensuring that administrative and operating procedures, which they would be preparing or overseeing preparation of, meet the requirements of the design bases.

From the beginning, the Station Staff took the lead role in preparing and updating administrative and operating procedures. Westinghouse and UE&C only provided reference material in accordance with their respective contracts. Procedure development work began in 1978. Commercial operation of Seabrook Station Unit 1, originally scheduled for 1979, was significantly delayed. However, in working to meet current schedules, PSNH had, by the early 1980s, filled a high fraction of approved Station Staff positions The construction completion delays experienced by the project provided the Station Staff several additional years to prepare and refine administrative and operating procedures. In general, procedures were scheduled to be approved at least six months prior to their anticipated use. By initial core loading in 1986, essentially all administrative and operating procedures required for plant operation had been initially issued; and many of these procedures had been revised based on trial and/or actual use.

Station Staff managers, supervisors, engineers and technicians in the Operations, Maintenance and Technical Support Departments all took an active role in the preparation, review and approval of the initial issue of administrative and operating procedures. In preparing and reviewing procedure drafts, responsible individuals consulted various reference documents for

applicable design basis requirements. Reference documents included the PSAR, the FSAR, NRC regulatory guides, NUREG-0896 (the SER for Seabrook Station) including all supplements, other NUREG documents as appropriate, the Seabrook Station Operating License and Technical Specifications. Documents containing the engineering design basis were also consulted to ensure completeness of information. These documents included system descriptions prepared by either Westinghouse or UE&C, vendor technical manuals, design drawings, equipment specifications, and official project lists such as the equipment and valve lists.

2. Initial Testing Program

The Startup Test Department (STD), a subgroup of YAEC, conducted the Station's initial testing program. Six engineers from the Station Staff were assigned full-time to the STD. The intent of this assignment was to ensure that a core group of Station Staff engineers would gain valuable knowledge and experience during the initial testing program which could be beneficially applied in the future. The initial testing program consisted of two major phases. The first phase, called *pre-operational testing*, began with checks, measurements and tests to verify proper construction, equipment installation and readiness for operation. Pre-operational testing continued by placing equipment and systems in operation for the first time and testing their performance against acceptance criteria derived from design basis requirements. The pre-operational testing phase was completed by conducting several, large-scale tests involving major portions of the Station in operation as an integrated unit. The second phase of the initial testing program was called *startup testing*. This phase began with preparations for initial reactor criticality, and was completed by conducting tests at various power levels up to and including full power. Tests conducted during the startup testing phase verified that structures, systems, components, and ultimately the Station as an integrated unit, performed according to their respective design bases.

Work done by the STD during the initial testing program significantly contributed to the development of the Station's operating procedures, and helped to ensure that they incorporated design basis requirements. The STD prepared procedures for the tests conducted during pre-operational and startup testing. For equipment operation and major plant evolutions conducted

during testing, the STD utilized the Station Staff's operating procedures to the maximum extent possible. In doing this, the STD verified the technical adequacy of the operating procedures and validated their usability. Exercising operating procedures during the initial testing program provided an early opportunity to identify and correct any deficiencies in the procedures could be identified and corrected at a very early opportunity.

In preparing for and conducting pre-operational and startup testing, members of the STD established a close working relationship with Westinghouse, UE&C and other on-site contractor personnel responsible for design and construction of the Station. This relationship helped the STD to become familiar with design basis requirements. The STD used this knowledge in their preparation of pre-operational and startup test procedures and in their review of operating procedures to be exercised during testing. During conduct and following completion of the initial testing program, pre-operational and startup test procedures and test results documentation provided a valuable source of reference material for incorporating design-related information into operating procedures.

The initial testing program also helped to provide an orderly transition conducted from construction to operation, in which authority over Station equipment was transferred from organizations responsible for design and construction to the Station Staff.

Following completion of the initial testing program, a majority of the originally assigned engineers and technicians joined the Station Staff. Many STD members were placed in key engineering, supervisory or management positions in the Station Staff.

3. <u>Emergency Operating Procedures (EOPs)</u>

The Emergency Operating Procedures (EOPs), which collectively consist of the Emergency Response Procedures (ERPs), Emergency Contingency Actions (ECAs) and Functional Restoration Procedures (FRPs) at Seabrook Station were developed in a manner which

effectively translated applicable design basis requirements into them[2]. Since their initial development, the EOPs have been and continue to be maintained in a manner which effectively preserves and updates applicable design basis requirements. A full-time EOP Coordinator ensures that EOPs and associated documentation are properly controlled and maintained.

The EOPs were developed in accordance with the Westinghouse Owner's Group (WOG) Emergency Response Guidelines (ERGs). These guidelines provided generic technical guidance applicable to all Westinghouse-designed commercial pressurized water reactor (PWR) nuclear power plants. The ERGs provide prioritized operator guidance for recovering the plant from an emergency transient while simultaneously ensuring that the plant safety state is explicitly monitored and maintained during recovery.

The ERGs are validated by a program which realistically and dynamically tests the ERG framework and written guidelines. Both the technical and human factors adequacy of the guidelines are tested using actual utility personnel on a real-time, full-scale simulator. Revision 1 of the ERGs was validated in late 1983 at the Seabrook Station Simulator by using Seabrook-specific EOPs which were based on and had been developed in parallel with the Revision 1 ERGs. Since that initial validation, the Seabrook Station EOPs have incorporated Revision 1B of the ERGs and have been maintained in accordance with the WOG ERG Maintenance Program.

To support maintenance and revision of the EOPs Seabrook-specific Technical Guidelines were developed. These Technical Guidelines document information necessary to integrate plant-specific information into the generic ERGs. For example, the Seabrook technical guidelines provide the technical basis to translate engineering data derived from transient and accident analyses into a basis for the Seabrook EOPs. The Seabrook Technical Guidelines also document on a step-by-step basis all deviations between the Seabrook EOPs and the corresponding ERGs. The explanation or basis for each deviation is documented.

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2 In this report, the term "Emergency Operating Procedures" or "EOPs" will, unless otherwise stated, be understood to refer collectively to EOPs, ERPs,ECAs and FRPs.

The generic ERGs require the input of plant-specific setpoints and other numerical values to produce the EOPs. The specific setpoints and other numerical values used in the Seabrook EOPs are formally documented in the Setpoint Study. In addition to setpoints and other numerical values, the Setpoint Study includes information pertaining to the source of each EOP value and a historical record of changes affecting each value.

Revisions to the EOPs are made in accordance with administrative controls contained in the Operations Management Manual (OPMM). These administrative controls include requirements for review, verification and validation of EOP revisions. EOP revisions are reviewed by the EOP Coordinator, the Operations Manager and the SORC. Verification of EOP revisions is required to ensure that the revisions do not create inconsistencies with EOP source documents. Validation of EOP revisions is required to ensure that the revised EOPs remain adequate both technically and from a human factors point of view. Validation methods for EOPs include tabletop discussions, operator walkdowns and real-time exercises on the simulator.

4. <u>The Role of the SORC</u>

Several years preceding issuance of the Station's first operating license (NPF-56), the Station Operation Review Committee (SORC) was formed and began holding meetings on a regular basis. As required by the Technical Specifications, the SORC is an interdisciplinary body. Convening the SORC at this time helped to ensure that it was effectively performing its function at the time of issuance of the initial operating license in 1986.

At an appropriate time prior to initial core loading, the SORC also began reviewing startup test procedures and design modifications. This illustrates the prudent steps taken by the Station Staff to ensure that, as the time for initial core loading approached, they obtained and maintained an up-to-date knowledge of the as-designed and operational status of Station equipment.

5. Design Change Impacts on Procedures

During construction, a large number of design changes were implemented. Many design changes physically modified structures, systems or components. Other design changes affected only documents. All design changes were initially implemented under the administrative control of UE&C. Design changes were initially documented as Engineering Change Authorizations (ECAs), Design Change Notices (DCNs) or Field Change Notices (FCNs). Design change documents were tracked by UE&C until the changes were completed and the related information was incorporated into official design and/or informational documents such as drawings, specifications, project lists, system descriptions and the FSAR. Station Staff and STD personnel reviewed design change documents as necessary to reasonably ensure that procedures reflected the latest design.

As discussed in more detail in the Response to Request (c), in 1987, NHY assumed responsibility for design control of Seabrook Station. To effectively carry out this responsibility, NHY developed and implemented a design control program. This program provided for preservation of the design basis of the Station, evaluation of proposed changes for unreviewed safety questions and, in coordination with other administrative programs, effective incorporation of design basis requirements into procedures and other official documents.

Over time, the details of the design control program have been revised and refined; but the basic elements have remained relatively unchanged. The current design control program is described in detail in the response to Request (a).

Several sections of the design control manual (DCM) address the design basis and translation of related requirements into procedures. For example, the narrative description of each proposed design change is required to include subsections entitled: "Basis of the Current Design," "Design Inputs/References," "Detailed Design," "Implementation Considerations," "Operational Considerations," and "Test Plan." In the Operational Considerations subsection, the design

engineer is required to identify how the proposed design change would modify operational, surveillance and maintenance practices or create new ones.

In current practice, affected Station Staff departments are notified during the design phase of each proposed design modification. This notification provides an opportunity for the affected department's representative to review the proposed design change for potential impact on procedures within the purview of that department. Additionally, design modifications which require a *safety evaluation* per 10 CFR 50.59 are reviewed by the SORC. These review opportunities help ensure that all affected departments review the proposed design modification and update any impacted procedures. Before permanently modified equipment can be released to the Operations Department for normal use, it must be declared "Operable." One of the requirements imposed by administrative procedures for declaration of Operability is confirmation from the Operations Department that applicable procedures have been updated to reflect the effects of the change(s).

6. Procedure Upgrades

Management considered the frequency of occurrence of personnel errors in 1993 to be unacceptably high. In 1993, the NRC also noted an increasing trend in personnel errors and other human performance deficiencies, and expressed the opinion that a comprehensive approach was required to address the problem.

In response, management formed a "Personnel Error Response Team" (PERT) to determine the underlying issues related to the personnel errors. One issue identified was overly complex processes, programs and procedures. To address this issue, North Atlantic initiated a procedure improvement program, which became known as the "Procedure Upgrade Program" (PUP). The purpose of the PUP was to simplify and improve the clarity of procedures with the ultimate goal of making procedures easier to use. The PUP was a substantial, resource-intensive initiative, which would be a multi-year effort resulting in the revision of approximately 3000 technical procedures. In 1994, the PUP had two major goals: 1) to reduce by consolidation the number of

manuals in the existing North Atlantic manuals program and to clarify and make more user-friendly their guidance; 2) to review and re-format technical procedures to improve accuracy, simplify, improve clarity and user-friendliness. Work on the PUP began in March 1995.

Requirements for the administration of procedures were consolidated into the North Atlantic Procedure Administration Manual (NAPA). The NAPA includes administrative guidance for translating design basis information into procedures. In its instructions for both preparation and review of procedures or revisions, the NAPA includes requirements to ensure that Technical Specifications and UFSAR requirements and other commitments to the NRC are properly addressed. New procedures and procedure revisions are prepared by subject matter experts. The NAPA defines a verification and validation (V&V) process by which new or upgraded procedures and revisions thereto are proven to be technically accurate. Independent review is a mandatory part of the V&V process. The independent review is conducted by a Station Qualified Reviewer (SQR) who is technically knowledgeable in the procedure subject matter, and is independent of the procedure preparation process. The independent reviewer is responsible for verifying the technical accuracy and completeness of the procedure. The independent reviewer is also responsible for determining safety evaluation applicability per 10 CFR 50.59; and, if necessary, performing a safety evaluation. Criteria for the independent reviewer includes specific questions designed to prompt the independent reviewer to consider many procedure attributes. Several of these questions address the adequacy and completeness of translation of design basis information into the procedure. For example, one question asks in essence: Has the procedure incorporated applicable requirements from the Technical Specifications, UFSAR, codes, standards, regulations and commitments?

In July 1996, while conducting a surveillance test with an upgraded procedure, a sequence of steps was identified which, if followed without question, could have caused a reactor trip. The reactor trip was avoided only because the technicians performing the surveillance test were attentive to detail and applied a questioning attitude. As a result of this near-miss event, management temporarily banned the use of all upgraded procedures which had not been fully exercised since their most recent issuance, revision or change. A validation process was

established to ensure that upgraded procedures are technically sound before removing the temporary ban on their use.

The root cause analysis performed in response to this near-miss event resulted in recommendations pertaining to the PUP itself. As a result, the upgrading of technical procedures has been temporarily suspended while the new manager, assigned in December 1996, re-evaluates the objectives, methods and priorities of the procedure upgrade effort.

The team which revises and upgrades procedures is now called the Procedures Business Group. Plans for the future activities of the Procedure Business Group are currently being made. The short and intermediate-term plan involve the following three elements: 1) address first quarter 1997 operational procedure needs, 2) address procedure needs to support the upcoming fifth refueling outage (ORO5), and 3) develop and obtain management approval for a revised procedure upgrade process. The long-term goal is to complete the procedure upgrade process on or as close as possible to its originally scheduled completion date.

7. Audits and Assessments

The team reviewed reports of audits, assessments, self-assessments and inspections performed by both internal and external groups/organizations issued between 1983 and 1996 that dealt with configuration management.

Included were assessments conducted by the Independent Review Team (IRT). The IRT was established in approximately 1984 as a management "check and balance" to facilitate construction completion and commencement of commercial operation. The IRT function has been retained in the operational phase to independently review activities and issues as requested by management. Organizationally, the IRT is now part of Nuclear Oversight.

The IRT reviews of power ascension and configuration control addressed configuration control and methods used to maintain knowledge of plant status. Numerous recommendations were

made to improved performance in these areas. The IRT concluded in their review of associated circuits concerns that programs and procedures used during the construction and startup phases properly implemented the associated circuits design concept and controlled design, procurement, installation and testing of associated circuits. For the operational phase, the IRT concluded that engineering programs and procedures provide adequate controls to ensure the associated circuits design continues to meet licensing commitments. In a 1993 report, the IRT assessed North Atlantic's programmatic controls for maintaining Current Licensing Basis (CLB) commitments. The IRT concluded that prior to issuance of Seabrook Station's full power operating license (OL), programmatic controls were not adequate but that commitment identification and tracking initiatives placed in effect following issuance of the full power OL addressed the program's weaknesses.

In June, 1989, an event occurred during the Natural Circulation Startup Test which led to an assessment of the startup testing program by reviewing selected startup test procedures. The assessment included review of the 10 CFR 50.59 evaluations performed for each startup test procedure, verification that the guidance in NRC Regulatory Guide 1.68 and related UFSAR commitments would be met and a review of the potential for unplanned trips or ESFAS actuation. The assessment concluded that the startup testing program verified that applicable design basis requirements were met.

Between 1990 and early 1996, Engineering, like most groups or departments within North Atlantic, conducts self-assessments. Engineering Self-Assessment Reports (ESARs) document pre-emptive, reactive and periodic self assessments performed by engineering to identify (or respond to) issues or adverse conditions and to find ways to improve performance. In preparing this response, the team reviewed a selected sample of twenty-two ESARs issued between 1990 and 1996. The sample included self assessments addressing design basis, UFSAR and configuration control topics. The team's summary report concluded that the sample ESARs did not indicate any programmatic deficiencies in engineering programs. Identified issues were either resolved or are being addressed. The sample of ESARs did not identify any issue explicitly related to translation of design basis requirements into procedures.

The team reviewed thirty-five NRC inspection reports issued between late 1989 and 1996. The sample of NRC inspection reports selected were engineering-related inspections, SALP reports, and any other inspections considered likely to contain information pertaining to design basis, configuration control or the corrective action program. The review included all Notices of Violation (NOV) issued between late 1989 and 1996.

The sample of inspection reports included several instances in which the design and licensing bases were not properly preserved in Station procedures. For example, in inspection report (NRCI) 96-10, a notice of violation (NOV) was issued because of a procedure change which permitted a configuration of the EFW System which was inconsistent with design basis requirements. Similarly, in NRCI 96-08, a NOV was issued because the EFW pumphouse floor drains were covered with duct tape (in accordance with an approved procedure) during painting in the area. A NOV was issued in NRCI 94-24 for a surveillance procedure which briefly permitted an equipment configuration outside the design basis. Our review of the NRC reports, however, has not identified a significant programmatic-level problem in the area of procedures properly reflecting design and licensing bases requirements.

In 1990, the NRC conducted a week-long team inspection of Emergency Operating Procedures (EOPs), Abnormal Operating Procedures (AOPs) and all procedures referenced in EOPs and AOPs. In their overall assessment, the NRC found that the program for generation and maintenance of the procedures was very good. No violations or deviations were noted.

The Institute of Nuclear Power Operations (INPO) conducts evaluations of nuclear power plant activities, makes determinations regarding plant safety and evaluates the effectiveness of management, programs, and technical activities. INPO evaluations do not specifically address configuration control. However, evaluations address activities necessary to establish and maintain the plant within the design bases. The team reviewed all five reports of INPO Evaluations conducted between September 1989 and May 1995. The review indicated that INPO identified areas related to configuration control, including procedures, which could be improved.

In the September 1989 and July 1992 evaluations, INPO noted areas in which procedures needed to be improved but, in general, considered them to be adequate. The procedure upgrade program discussed earlier in this response is addressing these areas.

Conclusion

The activities described above such as initial development of Station administrative and operating procedures, the initial testing program and mechanisms used to transfer authority and responsibility for the Station from engineering/construction organizations to the Station Staff and efforts to obtain the various operating licenses were collectively effective in establishing design basis requirements in procedures. From that point on, design basis requirements were reasonably well preserved and kept current by implementing North Atlantic's administrative programs, especially those associated with design change control. Operating experience resulted in discovery of a relatively small number of discrepancies between design basis requirements and procedural requirements. Appropriate corrective action was recommended and was, is being or will be appropriately dispositioned. Independent assessments including those associated with this response, have identified further areas for improvement. Those findings also have been or will be appropriately dispositioned.

Oversight activities conducted by both internal and external organizations have consistently concluded that Seabrook Station design requirements have been effectively incorporated into procedures although they have identified discrepancies which have been documented and dispositioned through the corrective action process.

The assessment of current processes and procedures, conducted as part of this response, evaluated the effectiveness of current processes, including the design change implementation process, and concluded that changes to the design bases were being translated into operating, maintenance and testing procedures. The assessment did, however, indicate that guidance on identifying and tracking Station procedures impacted by plant design modifications should be

strengthened. This observation had already been made in a common cause analysis conducted prior to this self-assessment. As discussed in Section VI, a number of specific steps to address this issue have already been improved for implementation.

The results of intensive SSFI-type, vertical slice reviews performed on seven risk-significant systems are reported in the Response to Request (d). Although discrepancies and areas for improvement were identified, the results do not indicate that there are any significant issues associated with translation of design basis requirements into procedures.

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

Summary

Seabrook Station's design bases are contained in the UFSAR. Those design bases are implemented and incorporated into the configuration of the plant's structures, systems and components through drawings, specifications, calculations and analyses. The 10 CFR 50.59 and design control processes are the primary tools for ensuring that plant changes, either physical, procedural or administrative, do not place the plant outside the licensed and approved design bases. The current configuration and design control program, including the 10 CFR 50.59 process, is described in the Response to Request (a).

The following are the basic elements of our rationale for concluding that systems, structures and component configuration and performance are consistent with the design bases:

- A comprehensive and accurate baseline for the design bases was initially established in the FSAR and in the implementing drawings, calculations, analyses and specifications. Extensive walkdowns and verifications were made to provide assurance that the plant was within the design bases at the time of construction turnover.
- Changes to the plant have been controlled by the 10 CFR 50.59 process and design change control processes.
- Audits and assessments conducted by North Atlantic, including the intensive configuration control assessments described in the Response to Request (e), those of third parties, and the NRC have consistently concluded that our programs have adequately maintained the plant within the design bases.

A discussion of each of these elements is provided below.

Discussion

1. Establishment of the baseline for the design bases.

a. FSAR

On June 29, 1981, as part of the application for Seabrook Station's operating license, PSNH tendered the Final Safety Analysis Report (FSAR). Utilizing the format of NUREG-75/094, the FSAR provided the information required by NRC Regulatory Guide 1.70, Revision 3. The NRC's safety review of the Seabrook FSAR was based on post-TMI detailed guidance of NUREG-0800 (Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors). In the nine years that followed before receipt of the full power operating license, the FSAR was amended 21 times. These amendments responded to requests from the NRC for additional information, and provided updated information to reflect changes to plant design, construction, organization, ownership, programs and procedures.

While PSNH was the plant's managing agent, procedures known as Project Policies for Seabrook Station governed design, construction, early equipment operation and testing, quality assurance and quality control. Project Policy (PP) 16 defined the method by which Westinghouse, United Engineers and Constructors (UE&C), Yankee Atomic Electric Company (YAEC) and PSNH originated and reviewed draft sections for the original issue of the FSAR. PP 16 provided a systematic and controlled method of issuing and reviewing draft FSAR sections while maintaining the accuracy and completeness of its information. The YAEC Seabrook Project Office coordinated the receipt, review and revision of FSAR draft sections under PP 16.

Project Policy 22 (Control of Design Changes) established a method for formally tracking and incorporating approved design changes into an FSAR amendment. PP 22 is important because it demonstrates that PSNH, as the utility, construction permit holder and prospective licensee, had

taken ownership of the FSAR and was engaged in systematically updating it as design changes were made.

During the interval between application for and receipt of the full-power operating license, numerous commitments were made to the NRC by PSNH and its authorized agents. The disposition of many included incorporation into an FSAR amendment. To ensure follow-through on these commitments, they were recorded and tracked on a computer data base called the Project Commitment Tracking System. Commitment priorities and requirements for use of the database were documented in PP 26. One priority was "Requires formal FSAR change." This attribute of PP 26 provided assurance that the appropriate FSAR updates were made.

During the last few years of construction, some proposed design changes were not immediately implemented but were deferred for further consideration following issuance of the operating license (OL). In 1984, PP 23 (Handling of Deferred Design Changes) established a formal method to document, track and transfer responsibility for deferred design change proposals to the Station's operating staff. Deferred design changes that the operating staff subsequently implemented went through the design change process. This ensured that the impact of the proposed design change on the FSAR and/or Technical Specifications was evaluated in accordance with 10 CFR 50.59 and that, when implemented, the FSAR was updated to reflect its effects.

In 1985, with construction nearly complete, New Hampshire Yankee (NHY) conducted a consistency review of the FSAR to ensure that it was internally consistent and reflected the latest design. FSAR sections were reviewed by experienced engineers with a good working knowledge of the material. The review included a verification that numerical values were accurate and traceable to a source document. Desi[illegible]ange documents generated between January 1, 1981 and April 30, 1985 were reviewed to ensure that, where appropriate, the FSAR was updated to reflect the effects of these changes. Other FSAR informational discrepancies were identified, documented and either corrected or tracked for future correction. Amendment 56 to the FSAR, submitted on November 26, 1985 incorporated the results of the consistency review.

In May, 1989 NHY received the low-power operating license for Seabrook Station. Emergency Plan litigation before the licensing boards, however, delayed full-power licensing. In order to ensure that the FSAR was kept current during this period, NHY developed two FSAR amendments[3] in the year between low-power and receipt of the full-power license in March, 1990.

The Updated FSAR was submitted in May, 1991. In addition to incorporating approved changes since the last amendment, a substantial review was conducted in preparation for this initial UFSAR submittal. A subject matter expert was designated Chapter Coordinator and placed in overall charge of each UFSAR chapter. The Chapter Coordinator reviewed and revised the text, tables and figures as necessary to ensure that the information was accurate, complete, up-to-date and in conformance with Regulatory Guide 1.70, Revision 3. When material required specialized expertise, the Chapter Coordinator delegated the review responsibility to experts at NHY, YAEC, Westinghouse or UE&C. The finalized information in each UFSAR chapter was evaluated per 10 CFR 50.59 and reviewed by the Station Operations Review Committee (SORC) before submittal.

b. Calculations, Specifications and Drawings

On July 8, 1987, nearly two years before receipt of the low-power operating license, New Hampshire Yankee assumed direct control of all site engineering and design activities from the Architect Engineer (UE&C). The smooth transition of the Station's design engineering organization from construction and startup to an operations-oriented mode was the culmination of a number of programs that had been initiated jointly by NHY and UE&C.

Over a six month period in 1983, a Seabrook Calculation Appraisal Team performed an initial close-out of safety-related calculations in preparation for the turnover of responsibility from

[3] Amendment 63 was submitted in April, 1990, about one month after full-power licensing

UE&C to NHY. As a result of the team's findings, a Safety-Related Calculation Close-out Program was initiated. This program served two functions critical to establishing a solid baseline for the subsequent turnover. It provided for a review that verified significant calculational assumptions and ensured that they were based on the latest source documents. It also included a study of the administrative aspects of preparing, revising and controlling calculations in order to assist NHY in developing the processes necessary to successfully take over these responsibilities. The program was completed in January, 1985.

When NHY assumed control of new drawing generation and revisions to existing drawings on July 8, 1987, it did so in accordance with the NHY Design Control Program and the NHY Engineering Procedures Manual. These programs controlled the development, updating, release and distribution of drawings. One of their major objectives was to ensure that drawings used by plant operators reflected the latest installed configuration. To facilitate this, Station drawings were broken down into three categories. Category 1 drawings are essential and directly support plant operations; Category 2 drawings do not directly affect unit operations but are required to support design and configuration management activities. Category 3 drawings are historical in nature and do not affect plant operations. Although the definitions and updating requirements for these categories have evolved, the basic philosophy that any Category 1 drawing must be formally updated before the system or component shown is deemed operable remains in effect.

As part of the preparations for turnover from the architect-engineer, NHY also converted the Station's manually drafted Piping and Instrumentation Drawings (P&IDs) to new Computer Aided Design based P&IDs. In so doing, NHY developed two new series of P&IDs which assisted operations and configuration management. The Series 'B' P&IDs are Category 1 and were prepared for direct support of operations. They represented an uncluttered drawing of primary and secondary system flow, eliminating unnecessary construction notes and details. The new Series 'D' P&IDs are Category 2 and included the information on the Series 'B' but also contain specific design information such as line numbers, isometric drawing numbers and boundaries, pump specifications, and tank capacities etc.

c. Walkdowns and Design Verifications

As part of construction turnover and the transition of design and configuration management responsibilities from the architect-engineer, a wide range of walkdown and design verifications were performed. These included:

- A walkdown of the P&IDs to verify their technical and operational representation.
- Piping System As-Constructed Verification Program. This program established the inspection requirements for field walkdowns which were performed to verify the final configuration of all ASME Code piping systems and pipe supports and selected non-code piping systems.
- Piping System Field Walkdowns. The field walkdowns verified that the as-constructed configuration of the piping and pipe supports were correct and within specified tolerances. The information obtained was used as input for the final configuration verification analysis and the final configuration reconciliation stress report.
- The Amplified Response Spectra Verification Program. This program provided the appropriate controls to verify that safety related structures, systems and components were designed in accordance with the correct Amplified Response Spectra.
- The Verification Program for Amplified Response Spectra for Supports for Seismically Analyzed Piping. This program provided a systematic review of seismically analyzed safety related and non-safety related piping in order to verify that the seismic effects were considered and applied as required.
- Beam Verification Program. This program provided verification that the design of Seismic Category I structural steel, which was initially designed based on conservative assumed loads, was adequate for the final system loads.
- Supplemental Steel Verification Program. This program established the methods to verify the adequacy of supplemental steel members and their connections for the loads imposed by electrical, piping, HVAC and I&C supports.

- Concrete Load Verification Program. This program provided the verification that walls and slabs of safety related structures, which were initially designed with estimated uniform loads, were adequate to withstand the final loading conditions.
- Piping Component Reconciliation with Design and Service Conditions. This program established the guidelines for the reconciliation between component design conditions or items such as valves, piping subassemblies and other ASME Code stamped components and the system design conditions.

In addition, at the conclusion of the construction phase and prior to the initial testing program, a Construction Completion Turnover Program was completed. Its purpose was to provide direction and requirements to the architect-engineer and the various site Contractors for the systematic turnover of Station systems to the Startup Test Department and for the review and transfer of construction records to the Station staff. It was executed around system boundary identification packages (BIPs) which defined a testable Station system or portion thereof by the identification of all equipment and components within the boundary of the BIP. The Construction Completion Turnover Program required the contractors to perform and document walkdown inspections of Station systems in order to verify that the system was complete and correct as specified by the contract drawings. The BIP system walkdown inspections were conducted on an engineering discipline basis. There were walkdowns for the mechanical system BIPs, mechanical services system BIPs, electrical system BIPs and I&C system BIPs.

2. <u>Controlling change</u>

Changes to Seabrook Station's structures, systems and components have been controlled and maintained within the design bases by the design change process and 10 CFR 50.59 applicability screenings and safety evaluations. By the time the initial ("zero-power") operating license was received in October, 1986, Seabrook Station began applying 10 CFR 50.59 to ensure that changes did not place the plant outside its design basis, create an unreviewed safety question or be inconsistent with its operating license. In June 1989, the Nuclear Management and

Resources Council of the Nuclear Safety Analysis Center (NSAC) issued Guideline No. 125 (Guidelines for 10 CFR 50.59 Safety Evaluations). NSAC-125, as it is known, provides guidelines for developing and improving 10 CFR 50.59 Safety Evaluations. Prepared by industry representatives and extensively reviewed by the industry and the NRC, it provides guidelines for defining thresholds for unreviewed safety questions, determining 10 CFR 50.59 applicability and performing 10 CFR 50.59 safety evaluations. NHY incorporated these guidelines into Seabrook Station's procedures in 1989. They have also been used in the 10 CFR 50.59 training program at Seabrook Station. The 10 CFR 50.59 procedure, now contained in the Station's Regulatory Compliance Manual, specifically requires that, in all cases, either the person performing the evaluation or the independent reviewer shall have successfully completed this training.

The NHY design control program was initiated in 1986. To maintain consistency and continuity, many of the architect-engineer's design and administrative practices were incorporated. An effort was also made, however, to identify and apply utility industry best design practices. By the time that NHY took over responsibility for all design activities in July, 1987, the NHY design control program was in place, documented in the NHY Design Control Manual (NYDC) and supported by the NHY Engineering Procedures Manual.

Although the NYDC was replaced in 1996 by the Northeast Utilities Common Design Control Manual (DCM), North Atlantic fully controlled Seabrook Station's design processes. As discussed in the detailed description of the process in the response to Request (a), the basic vehicles for design change are the Design Change Record (DCR) for large complex safety-related modifications and the Minor Modification (MMOD) for straightforward, minor changes to safety or non-safety related structures, systems and components. There are a number of features of Seabrook Station's program which have been important in maintaining the consistency of the system, structure and component configuration with the design bases and ensuring that the UFSAR reflects the as-built design. They include:

- UFSAR Change Requests are included as part of the DCR and are reviewed and approved with the DCR as a single action. When a design change becomes fully operable, the UFSAR change becomes effective.
- The 10 CFR 50.59 safety evaluation screenings and safety evaluations are also a part of the DCR. In addition, even the screening questions must be provided with a written basis for DCRs.
- MMODs and other lower tier design change documents are screened to determined if they require a 10 CFR 50.59 safety evaluation. If so, they are converted to a DCR. In this way, the comprehensive description, consideration of impacts and technical and management review required for a DCR is applied.
- Applicability screenings and safety evaluations performed in accordance with 10 CFR 50.59 have been monitored and reviewed by the Nuclear Safety Audit Review Committee (NSARC) since 1988.

North Atlantic has endeavored to identify weaknesses and areas requiring improvement through assessments and oversight provided by NSARC, the QA organization and by the Station's Independent Review Team. Third party assessments by INPO, the NRC and others have also been helpful in improving the design and configuration management program. Some of the significant assessments in this area are discussed below.

3. Audits and Assessments

The first major configuration control self-assessment began even before receipt of the full-power license. In 1990, a multi-discipline task force was established. It found that the programs were adequate and performing their function but that they needed to be simplified and consolidated. There was also, the task force found, a need for substantial improvement in 'attention-to-detail' and training. The Station has been continuously working to improve in this area through the STAR (Stop-Think-Act-Review) Program, the 1993 Personnel Error Response Team (PERT) and other efforts.

After a 1991 incident involving an isolated loss of configuration control on one component led to the radioactive contamination of the demineralized water system and parts of the Turbine Building, the Station's Independent Review Team (IRT) was asked to form another task force to identify areas of continued weakness and to provide recommendations for improvement. In the 1992 report, the task force concluded that the programs and procedures that comprise configuration management contained the necessary elements and were adequate to maintain the plant within its design bases. It criticized management, however, for not adequately implementing the recommendations of the 1990 task force. It also stated that management needed to be more effective in communicating a clear message to employees about its expectations with respect to configuration management. Another major finding was that the configuration control procedures were often too complex. An issue that is currently being addressed through the Procedures Upgrade Program.

The 1992 report recommended establishing a clear Station policy on configuration management, clear management expectations on this policy, communicating the policy and expectations to employees, and instituting strict accountability for each aspect of the program. Many of the recommendations contained in this report are people-oriented and must be continually reinforced. Additional work in the procedural elements is being addressed under the auspices of the procedure upgrade program.

Several more recent third party assessments have been conducted on the subjects of design and configuration management. As a result of the heightened industry-wide sensitivity to configuration management issues and the accuracy of the UFSAR in particular, North Atlantic asked YAEC to conduct a comprehensive assessment of configuration management and commitment control. The overall conclusion in their 1996 final report was that Seabrook Station has an effective program for managing the design bases. According to the report, the Station can thoroughly and accurately incorporate design changes and ensure that other impacted documents, such as the FSAR are appropriately reviewed and changed. The report also found a demonstrated ability to track and implement licensing commitments.

A subsequent report on change control by an outside consultant (Independent Advisory Services), sponsored by a joint owner oversight group representing owners not affiliated with NU, also concluded that the UFSAR reflects the as-built design although they found evidence of errors due to lack of attention to detail.

While NRC inspections have also identified individual discrepancies in maintaining consistency between the physical plant and the design bases, the most recent Systematic Assessment of Licensee Performance (SALP Report 96-99) report found that management fostered a strong safety focus and that engineering demonstrated a strong safety perspective as evidenced by thorough safety evaluations. In its April, 1996 report, the NRC's Integrated Performance Assessment Process team concluded that Seabrook Station has significant strengths in the areas of plant operations, maintenance, engineering, security, radiological controls and emergency planning. The IPAP also stated that the performance of design and system engineering was very good. With respect to the UFSAR, the IPAP found that the UFSAR wording was consistent with the observed plant practice, procedures and/or parameters for most situations. They did, however, cite several inconsistencies that highlighted the need for additional verification. North Atlantic management responded to this concern by heightening the Station's awareness of the need to ensure the accuracy of the UFSAR and urging that any inconsistencies be documented. A common cause analysis was then conducted on the Adverse Condition Reports filed on UFSAR inconsistencies. This analysis resulted in a number of detailed recommendations to address the concern. They have now been approved and are in the process of being implemented. The self-assessment that supports this response also focused heavily on the UFSAR. The results of each of the components of the self-assessment are addressed in the Response to Request (e). Additional future actions that we are taking to improve our configuration management documents and programs as a result of the findings are discussed in Section VI.

Conclusion

Seabrook Station took advantage of its long licensing process by ensuring that a thorough evaluation of the plant and its conformance to its design bases was undertaken prior to assuming responsibility for design control. Since assuming control in July, 1987, the UFSAR and the 10 CFR 50.59 process have been given a high priority. NSAC-125 guidelines were promptly incorporated into the safety evaluation procedure and training. UFSAR change requests have always been an integral part of a design change package allowing it to receive the same thorough review as the proposed modification. Audits and assessments, while finding isolated instances of inconsistencies in the past, have found the configuration management program adequate. It should also be noted, as discussed in Section VI, that North Atlantic had already undertaken several initiatives to improve configuration management. Section VI also describes the additional actions that we will take to address the findings of the assessment and reviews conducted specifically for this report.

D. 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.*

Several programs implement the processes described in the NRC's request. The central program, and the one that this response will focus on, is the Corrective Action Program (CAP) which uses the Adverse Condition Report (ACR) as its reporting mechanism. Two other programs that complement and can tie into the CAP will also be briefly discussed. They are the Operating Experience Review Program which provides a method for evaluating industry events and experiences and the Employee Concerns Program.

1. <u>Corrective Action Program</u>

The Operating Experience Program Manual (SSOE) provides the instructions and requirements for implementing a corrective action program in accordance with 10 CFR 50, Appendix B, Criterion XVI[4]. The SSOE provides for reporting, analyzing, assigning responsible managers, and the tracking of adverse condition report evaluations and corrective actions.

The CAP has been undergoing major changes over the past two years. Changes already made and changes in process are in recognition of the need to simplify programs that had become unnecessarily complex. The reason for these changes is that a simpler, more user-friendly process is more effective and more likely to be used. A simpler process also advances another major goal which is to lower the reporting threshold. A specific step we are taking to both simplify and lower the threshold is to incorporate lower tier reporting programs into the Station-wide Corrective Action Program, eliminate the lower tier reports, and use the ACR as the one

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[4] Criterion XVI states: Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

adverse condition reporting mechanism. During the transitional period, the lower tier programs will have specific criteria for determining when a condition meets the threshold of the corrective action program and should, therefore, be reported on an ACR.

The following paragraphs describe the ACR initiation, evaluation, review, closeout and trending process.

a. Who may initiate an ACR?

Anyone. A North Atlantic employee or a contractor may initiate an ACR. Management encourages all Station personnel to document any adverse condition on the ACR. The Station Director has made clear his expectation that Station personnel will not hesitate to submit an ACR where appropriate.

b. What may be reported?

An adverse condition is defined in the SSOE as a condition adverse to quality, or an unexpected or undesirable occurrence. A comprehensive, although not fully encompassing, set of examples of adverse conditions are provided in the SSOE. It is important to note, however, that the SSOE specifically directs that concerns about issues or conditions that are questionable, not obvious or indeterminate should be reported on an ACR. In other words, if in doubt, report. Once the ACR is submitted, the process ensures that the issue will be addressed.

The constant emphasis on the lowering of the reporting threshold has been reflected in the reporting statistics over the last several years. In 1994, there was a combined total of 359 SIRs and OIRs[5]. In 1995, after switching to the ACR report format early in that year, there were a total of 530 ACRs, as well as 9 SIRs and 28 OIRs. The total number of ACRs rose to 1470 in

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[5] The Station Information Report (SIR) and the Operational Information Report (OIR) were the predecessors of the ACR. The SIR was used for significant issues; the OIR was used for less significant events.

1996. The 1996 total reflects not only the lower threshold but the consolidation of some lower tier deficiency reporting documents in favor of the ACR.

c. Reportability and Operability Determinations

After an ACR is originated and signed by the originator's supervisor, it is brought to the Shift Manager. The Shift Manager is responsible for performing the initial operability and reportability assessment in accordance with 10 CFR parts 50.72 and 50.73 and documenting this on the ACR. The Shift Manager will notify Engineering and Technical Support if assistance in performing an operability determination is needed and Regulatory Compliance if additional reportability reviews are necessary. In any event, the Regulatory Compliance Supervisor will perform a final review to determine if a report is required in accordance with 10 CFR parts 50.72 and 50.73 as well as part 21, part 73.71, part 50.9 or other regulatory reporting requirements.

d. Management Review

A Management Review Team (MRT) was established in April, 1996. Its Chairman is the Assistant Station Director with the Station Director serving as the Alternate Chairman. All managers who report to the Station Director are members. The Licensing Manager ensures that any documented adverse conditions are reviewed for reportability requirements and prepares any necessary reports for MRT and SORC review prior to final submittal to the regulatory agency. The Nuclear Safety Engineering Manager, although not an MRT member, brings industry operating experience information and/or reports to the attention of the MRT to support the ACR review process. Other managers on the MRT are from maintenance, engineering, technical support, operations, quality assurance and control, planning and scheduling, procurement, site services, security, health and safety, and licensing. The MRT charter and responsibilities are provided in the Operating Experience Manual (SSOE).

The MRT meets daily to review all ACRs. They perform an initial review in which they determine significance and priority, assign a responsible manager, and specify the appropriate

disposition (e.g. root cause, apparent cause, corrective action, trending etc.). The MRT also performs a final review of completed high significance (Level A) and moderate significance (Level B) ACR evaluations. In the final review, they determine the adequacy of the cause evaluation and the corrective action.

The MRT is the successor to a review committee comprised of lower level management. This higher level of review was made feasible by establishing a hierarchy of ACR significance levels and allowing the MRT to focus on the conditions with higher significance. The MRT concept has had positive results that include the following:

- MRT managers obtain a broader picture of the corrective action program and are more readily able to discern adverse trends.
- MRT managers now more clearly recognize that adverse conditions reported in other disciplines may occur in their discipline if corrective actions to prevent recurrence are not global.
- The MRT has improved communications among managers and has facilitated the recognition and mutual resolution of adverse conditions.

The MRT forwards all Significance Level A ACRs to the Station Operating Review Committee (SORC) for review and final approval. Certain Level B ACRs meeting criteria delineated in the SSOE will also be sent to the SORC. Included among the Level B ACRs that go to the SORC are repetitive or similar adverse conditions that have a significant impact on Station operations, equipment or personal safety.

e. Significance Level

ACRs can be assigned one of three significance levels as follows:

Level A—High Significance—adverse conditions which have or potentially have serious safety consequences, radiation exposure, death, damage, financial impact, and serious discredit to the organization. A high significance condition reportable to the NRC in accordance with 10 CFR 50.72, 50.73 or 21 would also be an example of Level A ACR.

Level B—Moderate Significance—adverse conditions which represent recurring programmatic deficiencies, generic problems, adverse trends, and other problems which, if not corrected, can be expected to result in a Level A adverse condition in the future.

Level C—Low Significance—conditions adverse to quality which require the lowest level of response and require corrective actions, when appropriate, and trending to support performance assessment.

Significance Levels are assigned by the MRT and all Significance Level A and B ACR evaluations including proposed corrective actions are reviewed by the MRT. For Level A and B ACRs, the evaluator must consult with the ACR originator and the originator must review the final ACR disposition to ensure that the report is factual and correct and that the corrective actions could be reasonably expected to preclude future problems.

f. Extent of Condition

The MRT will review the ACR to determine the appropriate disposition. In addition to the operability determination that must be performed when there is a reasonable question as to whether the structure, system or component can perform its required safety function, the MRT may require one or more of the following:

- Event Evaluation—An event evaluation must be completed before operations are resumed after a reactor trip or engineered safety feature (ESF) actuation. It consists of a description of

the event including a chronology, its initiating conditions, a summary of operator actions, and corrective actions that must be completed before restart.

- Cause and Failure Analysis—This is an analysis of the apparent cause of an off-normal condition or unexpected response. It would be typically used for sudden component degradation of unknown origin, abnormal component behavior or repetitive but apparently dissimilar problems causing excessive maintenance. Where a more formal analysis is required, a root cause analysis would be prescribed.
- Root Cause Analysis—This analysis must be performed for a reactor trip or ESF actuation and may be prescribed for other Level A or B ACRs. It is a rigorous methodology to determine the underlying cause of the problem and to prevent its recurrence.

Although each adverse condition is normally reported separately to support trending requirements, a single evaluation may be done for a group of ACRs representing similar occurrences. This ensures that peripheral and underlying generic causes are identified and corrected and that the broader concerns have been addressed and corrected. It is also important to note that it is the expectation of the Station Director, as provided in the SSOE, that regardless of the minimum application thresholds specified above, these methodologies may be applied whenever the responsible manager deems it necessary.

g. Trending

Once the adverse condition is identified, the Station Director's staff assigns a preliminary Significance Level and priority and reviews past corrective action documents to determine if the condition is similar or repetitive. A repetitive or similar occurrence may have the significance level and/or priority raised to ensure that the underlying cause of the problem is addressed. If documentation indicates that low significance ACRs are developing an adverse trend, a separate ACR, with a higher significance level assigned, may be generated to evaluate the trend.

In order to ensure consistency and accuracy, the Quality Assurance Department is assigned responsibility for trending adverse conditions and providing ACR activity summaries. Quality Assurance provides a weekly summary report which is reviewed by the MRT and other managers and supervisors. Quality Assurance also provides monthly and quarterly trending reports. The quarterly trend report is presented and discussed at a scheduled Station Director's Performance Improvement Meeting. The Station Director may require responsible managers to take appropriate action, as necessary, based on the information provided at this meeting.

2. Audits and Assessments

A review of past self-assessments as well as NRC and third party inspections and assessments of the CAP reflect a program that has had growing pains but that has improved steadily. For example, in a 1993 Resident Inspector's Report (93-13), North Atlantic was cited for a lack of follow-up on corrective actions. This led to the formation of the Occurrence Review Committee (ORC), the predecessor to the MRT, which helped provide improved monitoring of deficiency reports and their corrective actions.

A 1994 North Atlantic audit revealed concerns related to timeliness, reporting threshold and complexity. That same year, the NRC cited North Atlantic for two violations related to the corrective action process. The first was for not promptly identifying an adverse condition in the Main Steam Isolation Valves (NRC Resident Inspector's Report 94-03); the second for slow resolution of a design issue with the emergency bus loss of voltage protection circuits (NRC Resident Inspector's Report 94-24). This led to a restructuring of the ORC to give it more technical expertise. Steps were also taken to lower the deficiency reporting threshold.

In April and May of 1994, an NRC Independent Oversight and Self-Assessment Team conducted a performance-based inspection of the self-assessment program at Seabrook Station. It included selected examinations of the effectiveness of the corrective action program in resolving deficiencies identified by Oversight and the line organization. Their June, 1994 report noted the recent initiatives that North Atlantic had taken to improve its program and found them to be on

target. They stated, however, that they had not been in place long enough for the NRC to conduct a full assessment. In 1995, a Joint Utility Management Audit (JUMA) audit cited substantial improvement in the corrective action program which they had considered fragmented and inconsistent in a report three years earlier. By April, 1996, the NRC's Individual Performance Assessment Plan (IPAP) team inspection found that, overall, the Station was effective in identifying problems and usually effective in resolving them although there were still some recurring problems, and timeliness was still an issue. The 1996 SALP report noted that the corrective action program continued to develop and mature.

3. Operating Experience Review Program

The Operating Experience Review Program (OERP) provides a method for evaluating selected regulatory correspondence, industry operating experience and Seabrook Station operating experience for its impact on the Station. It also provides for the dissemination of this information with appropriate recommendations to North Atlantic management. The Nuclear Safety Engineering Group (NSEG) manages the Operating Experience Review Program although the NSEG Supervisor may assign evaluations to other organizations.

Some of the documents reviewed through the OERP include the following:

- INPO documents
- NRC Information Notices
- Vendor documents including General Electric Technical Information Letters, Westinghouse Technical Bulletins, Westinghouse Vendor Letters, Industry 10 CFR Part 21 Reports, and other vendor related correspondence
- Adverse Condition Reports and Control Room Journal reviews

The NSEG performs the initial screening and review of operating experience documents to determine which require additional assessment. Those with the potential to have a significant

impact on the safety and reliability of the plant receive a formal evaluation. The NSEG Supervisor researches industry experience related to an ACR to identify similar industry events when deemed necessary. The NSEG Supervisor also reviews ACRs to identify events that may be of interest to other nuclear stations. If appropriate, they are shared with the industry via the INPO Nuclear Network.

4. Employee Concerns Program

a. Policy on Reporting of Concerns

North Atlantic recognizes that free and open expression of concerns is a fundamental characteristic of active communication and is essential for safe and efficient operations.

A concern in the context of this program is a statement expressed by an individual regarding Seabrook Station activities which may involve nuclear quality/safety-related or nonsafety-related hardware, documentation, management/personnel, or non quality-related management concerns such as cost, scheduling, etc. The two types of concerns can be defined as follows:

- Safety-Related/Quality Concern—A technical concern which could affect quality, and/or the safe and reliable operation of the nuclear facility.
- Nonsafety-related Concern—A concern which affects personnel safety, and those activities that are not directly related to the safe and reliable operation of the nuclear facility.

The Company maintains an open-door policy allowing any employee access to management for the reporting of a concern. This is the preferred avenue. Concerns reported to immediate supervisors or within the chain of command allow responsible management to receive first-hand knowledge of existing or perceived problems that require their attention for validation, appropriate action and information feedback.

The Company has established the Employee Concerns Resolution Program as a supplemental avenue for the receipt of employee concerns. It is recognized that circumstances may exist where an individual may not choose the company preferred path to lodge a concern. Regardless of the option chosen to report a concern, it is Company policy that protection be afforded the individual from reprisals, reprimands, harassment, intimidation, retaliation or criticism. Confidentiality, as appropriate, is afforded each individual with the assurance that the concern will be investigated and satisfactorily resolved.

b. Goals of the Employee Concerns Program

The goals of the program are as follows:

- Establish credible avenues for all employees to confidentially register their concerns without fear of discriminatory actions.
- Provide responsible management with sufficient information to take ownership of concerns, real or perceived, and initiate resolutions.
- Effectively investigate and resolve concerns followed by results feedback to create an open atmosphere of communication and trust.
- Maintain sufficient records to allow audit evaluation for program implementation and improvement.
- Initiate corrective action, when needed, with adequate evaluation to cause, and appropriate changes to avert recurrence of the concern.

At Seabrook Station supervisors and workers are trained to create and maintain an environment in which all employees are free to raise issues and or concerns. Employees are expected to raise and to respond to the issue/concern in a timely and constructive way.

Questions, issues and concerns are raised every day. The majority of these are raised and resolved in the ordinary course of our work using the Station's Corrective Action Program. The

Employee Concerns Program provides an alternative method for raising concerns. Employees (including contractors) are informed of the importance of raising safety concerns and how to raise them through normal processes, the Employee Concerns Program and directly to the NRC. They attend training programs to ensure they understand their rights and know that they have legal protection against discrimination. This is accomplished so that each employee fully understands his/her responsibilities for fostering a workplace environment that welcomes concerns.

Employees and contractors leaving Seabrook Station are given the opportunity to voice concerns at an exit interview with the Manager of the Employee Concerns Program. This, of course, includes contractors who were involved in the review which is the subject of this report. Those exiting outside the programs office hours are sent a letter soliciting any concerns or constructive comments they may have.

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

Summary and Overall Conclusion

As described in the responses to Requests (a), (b), (c) and (d), Seabrook Station's historical performance, its record of audits and assessments and its current programs, processes and procedures provide reasonable assurance that the plant has been, and continues to be, maintained consistent with the design bases and that deviations are reconciled in a timely manner. The results of the self-assessment described in this response support that conclusion. As with any examination of this scope and intensity, discrepancies and areas warranting additional attention and improvement have been identified. They do not, however, represent a programmatic breakdown. All ACRs generated by the self-assessment will be individually and collectively evaluated to ensure that potential programmatic weaknesses are addressed. Other actions that we are taking to follow-up on the self-assessment and fully address its findings are described in Section VI.

Seabrook Station began operations with a verified design bases and an established configuration and design control program, managed and implemented by the utility. That program has been monitored and assessed over the years with problems corrected, and weaknesses identified and addressed. Our conclusion that the plant is consistent with the design bases, therefore, rests not just on the effectiveness of current processes and programs but on the adequacy of the original design bases and the controls that have been in place since we assumed responsibility for design from the architect-engineer.

Being one of the most recent licensees was an advantage in that Seabrook Station was required to prepare an FSAR that could meet the rigorous acceptance criteria of the NRC's post-TMI Standard Review Plan (NUREG 0800). From initial docketing in 1981 on, changes to the FSAR were tightly controlled. Before receipt of the initial operating license, NHY performed a comprehensive consistency review. After licensing, but before submittal of the first Updated

FSAR (UFSAR), another review was conducted. Since then, UFSAR changes have been incorporated into the design control process, reviewed and approved with the change package and deemed effective when the change becomes operable. In this way, it does not constantly have to 'catch up' to design.

Plant Technical Specifications also underwent a series of reviews before licensing. These included reviews conducted under the auspices of the Technical Specification Improvement Program, audits of Westinghouse and UE&C analyses and line-by-line reviews both after low power and full power licensing.

The Construction Completion Turnover program ensured that acceptance of plant and systems from the architect-engineer was done in a systematic and thorough manner. It included extensive verifications and walkdowns to verify that the plant's structures, systems and components met the design requirements. After extensive reviews and audits of calculations, specifications and drawings, NHY assumed responsibility for design engineering from the architect-engineer in July, 1987, almost three years before receipt of the full power license. This allowed our own design and configuration management programs to be firmly established before full power licensing.

Over the operating life of the plant, changes to the plant's configuration have been controlled through the Station's design control program and the 10 CFR 50.59 process. The design control program that is in place today is described in the Response to Request (a). Based to some extent on the architect-engineer's program, it was enhanced by applying lessons learned from a review of industry best practices. It is protective of the design bases by requiring that plant design changes include a review of potentially impacted documents. It also requires that the 10 CFR 50.59 evaluation as well as the UFSAR change request be included with the change package. The 10 CFR 50.59 process itself has been based on NSAC-125 guidelines since 1989. The quality of 10 CFR 50.59 evaluations is monitored closely by the Nuclear Safety Audit Review Committee, a group of senior managers and outside technical experts. There have been instances

of non- or inadequate compliance with 10 CFR 50.59 that have been self-identified and/or cited by the NRC but overall the program has been effective.

North Atlantic and third-party audits and assessments have indicated that configuration management at Seabrook Station has been adequate. Since 1990, these assessments include two configuration control task force reports, an external configuration management assessment, several Joint Owner oversight reports as well as numerous quality assurance audits. The NRC, in their inspections and SALP reports, have noted individual configuration control deficiencies but have not found programmatic configuration control problems. In the 1996 SALP, the NRC rated Seabrook Station as Category 1 in Plant Operations, Engineering and Plant Support and Category 2 in Maintenance. The report observed that well-developed engineering programs and procedures were evident and that comprehensive safety evaluations demonstrated a strong Engineering safety perspective.

We believe that Seabrook Station's operating and regulatory record and the results of past audits, assessments and evaluations provide a basis for concluding that the Station's configuration control program has been effective in maintaining the plant within its design and engineering design bases. Nonetheless, we decided to conduct a comprehensive self-assessment that provides additional objective evidence for our conclusion. We did this to identify weaknesses and areas requiring improvement in our programs. Along with the entire nuclear industry, North Atlantic has become increasingly aware in recent years of the need to continually upgrade its configuration management and corrective action programs. In fact, a number of efforts had already been started, independent of the NRC's 10 CFR 50.54(f) request. These are reported on in Section VI.

The self-assessment constituted a serious commitment of time, talent and resources. Conducted by a team of highly qualified and experienced engineers, they consumed over 15,000 person-hours during the two and one-half month period that they were conducted. The vertical slice reviews alone consumed about 6,500 person-hours. The total scope of the various components of the self-assessment and their results provide a firm basis on which to support our conclusion that

the plant is being maintained consistent with its design basis. The overall conclusions of each component of the self-assessment is discussed below.

Results of the Self-Assessment Conducted for this Report

The following tests of the effectiveness of design and configuration control programs were conducted in accordance with the methodology described in Section III:

- An assessment of the current programs, processes and procedures
- Vertical slice reviews of seven risk-significant systems
- Assessments of nine engineering technical programs and topical areas
- A review of UFSAR Chapter 15 analyses
- A Technical Specification Review

A summary of the scope and overall results of each review and assessment is provided below. Detailed reports on the reviews and assessments and on the findings have been incorporated into the Engineering Self-Assessment Report.

All findings and recommendations generated by the teams are being dispositioned in accordance with the Stations' corrective action and design control programs. Findings that constitute an adverse condition, as defined in the corrective action program, are dispositioned on an Adverse Condition Report (ACR). As described in the Response to Request (d) above, one of three Significance Levels, A, B or C, in decreasing order of significance, are assigned to all ACRs by the Management Review Team. Findings, recommendations or questions that do not require an ACR are being tracked as on an Engineering Work Request (EWR), a Procedure Change Request form, or by being inputted directly into the Station's action item tracking system.

The ACRs have also been reviewed for reportability in accordance with 10 CFR 50.72 and 10 CFR 50.73. As a result, five Licensee Event Reports are being submitted.

a. <u>Current Programs, Processes and Procedures Assessment</u>

Description and Scope:

The main objective of this assessment was to determine if the links within the various procedures and procedure steps that constitute each of the assessed processes was adequate, i.e. do they have appropriate 'hand-shakes' and 'hand-offs.' The following five specific engineering design and configuration control processes were assessed:

- The 10 CFR 50.59 safety evaluation process
- The 10 CFR 50.71(e) FSAR update process
- The design change implementation process
- The corrective action program
- 10 CFR 50 Appendix B quality assurance implementation

Although the team looked at a total of 299 program, process and procedure links, it did not test the actual success of the processes in meeting their objectives—that was done primarily through the other assessments, especially the vertical slice reviews.

Conclusions:

The overall conclusion of this review was that the current programs, processes, and procedures are sufficient to operate and maintain Seabrook Station within the design bases. Several weaknesses and areas needing improvement were identified by the team. According to the review team, the guidance provided for determining the impact of design changes on procedures should be enhanced. The Approved UFSAR Change Reference Manual (AUCR) should be more widely available. Guidance on the requirement to perform 10 CFR 50.59 evaluations for design changes that have been declared partially operable should be enhanced. There are also a number of improper cross-references in the procedures. All of the team's findings have been appropriately documented and will be reviewed, dispositioned and corrected as appropriate.

b. **Vertical slice reviews**

North Atlantic conducted vertical slice reviews on the following seven systems:

- 125 VDC System
- Primary Component Cooling Water System
- Emergency AC System
- Emergency Diesel Generator System
- Emergency Feedwater System
- Plant Protection System
- Residual Heat Removal System

These are all risk significant systems that cover a wide cross section of engineering disciplines.

The reviews were conducted by seven system teams and one support team comprised of senior experienced personnel from North Atlantic, YAEC, and MPR Associates, Inc. (Alexandria, VA). The reviews were managed by MPR Associates, an independent engineering company with no prior involvement in the Seabrook project. About 6,500 person-hours were expended on these Seabrook reviews, which began in mid-November 1996 and extended into January 1997.

The reviews focused on identification of licensing and design bases for these systems and verification that these are accurately and consistently documented and effectively translated into the design, the physical plant configuration and the operating, maintenance and testing procedures of the systems. The reviews also assessed the effectiveness of plant processes, programs and procedures in maintaining the system design, configuration and procedures in conformance with the licensing and design bases. Effectiveness of corrective action was also evaluated, where applicable.

The reviews were based on the 27 topical areas listed in Table 1. These categories align with the information assessed in a typical Safety System Functional Inspection (SSFI) as described in NRC inspection procedure 93801. A set of attributes for each individual vertical slice system was developed and evaluated against these topical areas. An attribute for the Emergency Feedwater System, for example, is pump design including required head and flow, suction and discharge line-up. The topical areas checked for these attributes would be the calculations, drawings and specifications for the pumps; the UFSAR and design basis document description; NRC generic correspondence that impacts these attributes; operating, testing and maintenance procedures that refer to, or rely on, these pump design characteristics, etc. Not all topical areas are necessarily applicable to each attribute. The project did endeavor, however, to ensure that over the course of the vertical slice reviews, sufficient checks were made against each topical area to justify a reasonable conclusion on the adequacy of its implementation in the design process. Figure 3 is an example of the matrix form used to keep track of the system attributes selected for a vertical slice review and the topical areas against which they were assessed.

Over 3000 attributes of the seven systems, were evaluated against the 27 topical areas by the teams. The system attributes were selected for evaluation by each team based on their importance to the safety functions and design bases of the system. The specific topical area selected for review for each system attribute depended on the nature of the attribute and the importance of the information to validation of compliance of the system attribute with the system design basis.

The results of this vertical slice review provide additional assurance that the Seabrook configuration management program and its implementation have been effective in maintaining the plant within its design bases since its turnover to North Atlantic in 1987. The following overall conclusions are drawn from the vertical slice reviews:

- No programmatic breakdowns in the configuration control program were identified.
- Operating, maintenance and testing procedures and vendor information were found to be generally in conformance with the system and component design bases.
- The corrective actions program investigated in the reviews were found to be generally effective, and no instances of problem recurrence were identified.
- No major discrepancies were observed in the physical plant during walkdowns.
- Design basis and licensing documentation was found to be generally accurate and complete.

Areas of relative weaknesses were found in the Calculations and the UFSAR categories. The UFSAR discrepancies involved inconsistencies between descriptions in the UFSAR and the actual plant configuration and inconsistencies in statements made in different sections of the UFSAR or between the UFSAR and other design documents. These discrepancies indicate improvement is warranted in the processes for ensuring that the UFSAR is accurate and consistent with the plant configuration. Improvement is also indicated in the processes for ensuring that all factors affecting Calculation inputs, assumptions and methodology are considered.

The discrepancies identified in the UFSAR and in Calculations are being dispositioned under the analysis and resolution requirements of the ACR process. Actions and commitments that North Atlantic is making to improve performance in these areas are discussed in Section VI.

A description and scope of the review for each of the seven systems is presented in Table 2.

c. **Engineering Technical Programs Assessment**

Reviews of the following nine engineering technical programs and topical areas were conducted:

Programs

- Environmental Qualification Program
- Fire Protection Program (10 CFR 50 Appendix R)
- Testing Programs (Inservice and 10 CFR 50 Appendix J)
- Inservice Inspection Program (ASME Section XI)

Topical Areas

- Electrical Separation
- Post Accident Monitoring (Regulatory Guide 1.97)
- Individual Plant Examination of External Events (IPEEE)
- High Energy Line Break (HELB)
- Station Blackout (SBO)

These programs were selected because of their importance and because they have technical and administrative requirements that impact many different systems. They also cover a wide cross section of engineering disciplines.

The methods for the Engineering Technical Program reviews were individually structured and were conducted by nine program teams comprised of senior experienced personnel from North Atlantic, YAEC, and several consultant companies. Over 3,100 staff-hours were expended on these reviews, which began in late November 1996 and extended into January 1997.

The reviews focused on ensuring that the licensing and design bases for these programs are accurately and consistently documented and effectively translated into the design, the physical plant configuration and the operating, maintenance and testing procedures. The reviews assessed

the effectiveness of plant processes, programs and procedures in maintaining the design, configuration and procedures in conformance with the licensing and design bases. Effectiveness of corrective action was also evaluated, where applicable.

The results of these program reviews provide adequate assurance that the Seabrook configuration management program and its implementation have been effective in maintaining the plant within its design bases since its turnover to North Atlantic in 1987. The following overall conclusions are drawn from the engineering program reviews:

- No programmatic breakdowns in the review were identified.
- Operating, maintenance and surveillance procedures were found to be generally in conformance with the design bases of the programs and topical areas.
- No major discrepancies were observed in the physical plant during walkdowns.
- Design basis and licensing documentation was found to be generally accurate and complete.

The results also indicate that the programs, such as Environmental Qualification and Fire Protection were generally better understood and more consistently implemented than the topical areas such as Station Blackout or High Energy Line Break. The apparent reason for this is that the programs, because they impact the Station visibly and routinely, have a defined organizational structure within the Engineering Department. The topical areas have had a major impact on design but they do not impact the plant on as routine a basis as the programs. Their ownership over the years has, therefore, become more diffused within the Engineering Department. As a result of the knowledge gained from the self-assessment, North Atlantic will develop a plan to provide more structure and visibility to the topical areas.

The description of each program and the scope of the reviews are provided in Table 3.

d. **UFSAR Chapter 15 Review**

Description and Scope:

As a measure of the overall effectiveness of the configuration and design control processes at Seabrook Station, an assessment of UFSAR Chapter 15 was performed. Chapter 15 of the UFSAR was selected for the review because it is a focal point of the UFSAR that provides a significant analytical basis for the design of the plant. The objective of the assessment was to determine if the UFSAR has been maintained as an accurate representation of the design basis of the plant.

The assessment was supported by Westinghouse Electric Company, which was responsible for the LOCA analysis, and by Yankee Atomic Electric Company (YAEC), Nuclear Services Division, the company responsible for the remaining Chapter 15 safety analyses, including the radiological analyses. The support entailed providing access to the calculations, and explaining the decisions on inputs, methods and results which were not evident from the documentation.

An integrated assessment of the UFSAR Chapter 15 supporting documentation was performed to verify that the format and content are consistent with the recommendations appearing in regulatory guidance; to assure that the input documents, such as Topical Reports, Analysis Files, and Safety Evaluation Reports, are consistent with the information appearing in the UFSAR; and to verify that output documents, such as the Core Operating Limits Report, are consistent with the assumptions and limits established by the UFSAR analyses. Potential Impacts (PIs) were prepared for the analyses in Chapter 15 of the UFSAR from the review. The PIs were used in conducting the integrated assessment of the key attributes having an impact on the design basis documentation. While the assessment did not perform calculations, the adequacy of the assumptions and the reasonableness of the results were evaluated.

Conclusions:

As a result of the assessment, the following conclusions were reached:

- The transient and accident analyses in Chapter 15 of the UFSAR followed the guidance of Regulatory Guide 1.70 and NUREG-0800, including the level of detail in the documentation
- The input and assumptions and results are, for the most part, accurately described in the UFSAR. However, the equipment availability and performance characteristics provided do not appear to contain, in an unambiguous form, all of the parameters which have an impact on output documents.
- The methods used in the UFSAR were appropriate and licensed. The analytical techniques employed in the analyses reflect state-of-the-art methodology, most developed by industry groups such as the Electric Power Research Institute, and previously applied successfully to other plants.
- With the exception of UFSAR discrepancies noted in the assessment report, the system performance criteria required for the accident analysis, such as flow rates, response times, flow paths, are accurately described in the calculations and have been correctly transcribed in the UFSAR. The assessment concluded that the discrepancies are not significant.
- The NUREG-0800 acceptance criteria are satisfied or deviations are documented in the UFSAR or approved in an NRC Safety Evaluation Report.
- The results of the Safety Analysis have been appropriately incorporated into the Core Operating Limits Report (COLR).
- With the exception of UFSAR discrepancies noted in the assessment report, the appropriate input and assumptions have been correctly transcribed into 1- NHY - 250002, *Accident Analysis Input Parameters*, Revision 2, from the information provided in the UFSAR. The assessment concluded that the discrepancies are not significant.

All discrepancies and recommendations for improvement have been documented either on an ACR, an EWR or inputted directly into the Station's Action Item tracking system for evaluation and dispositioning. In addition, the UFSAR discrepancies noted above will be combined with

the UFSAR discrepancies identified in the other assessments and collectively evaluated through a common cause analysis. UFSAR Chapter 15 will also be re-assessed during the comprehensive UFSAR review. The UFSAR common cause analysis and the comprehensive review are also discussed in Section VI.

e. Technical Specification (TS) Review

The TS review consisted of the following two parts:

- Confirmation of appropriate numbers for selected numerical values in the TS, and
- Confirmation that the LCOs accurately describe the required system, and that the current system designs are consistent with the LCOs.

Background:

A TS verification project was initiated in 1986 to confirm that the initial Seabrook TS contained appropriate and accurate numerical values. The project team worked with United Engineers & Constructors (UE&C), Yankee Atomic Electric Company (YAEC), and Westinghouse, to validate the numbers. Again in 1992, a team from Seabrook met with Westinghouse personnel to review the bases that supported the Westinghouse supplied numerical values included in the Seabrook TS.

The TS data accumulated during the earlier TS verification project were tracked on a matrix which served as the platform for the review.

Scope and Description of Numerical Value Review

The first step of this review was to identify those TS values which either were not referenced to a basis, or were referenced to a basis which appeared questionable. About 160 numerical values fell into this category. About 50 were within Westinghouse scope and about 40 within YAEC scope. The remainder were reviewed by North Atlantic.

The bases for the Westinghouse- and YAEC-supplied values were obtained. For those reviewed by North Atlantic, a determination was first made as to whether the value was based on a calculation or other supporting document. The TS numerical value was then verified either

through review of the document or confirmation by a cognizant engineer. If the basis for a numerical value was the Standard Technical Specifications, it was reviewed for reasonableness.

Scope and Description of the Limiting Condition for Operation (LCO) Review:

In this review, Safety Limits and Limiting Safety System Setting (SL and LSSS) (Technical Specification Section 2), and Limiting Conditions for Operation (LCO) (Technical Specification Section 3) were reviewed against design basis documents, regulatory requirements, and the UFSAR, to confirm that the TS accurately reflects plant systems, and that current system design is consistent with the TS.

Each SL and LSSS, and LCO was reviewed to determine that:

- The bases for the requirements in the TS are accurate,
- The TS reflects the assumptions used in the accident analysis, and
- The TS and operational requirements reflect the capabilities of the systems as designed.

The results of the individual TS reviews were evaluated and a determination made as to whether each SL and LSSS, and LCO is appropriate for Seabrook Station, and whether it is consistent with the current design basis.

The review did not include Action Statement completion times, Surveillance frequencies, the appropriateness of Action Statements and Surveillances, the consistency of Surveillances with required system design capabilities, or a verification that the requirements of the TS have been translated into Station operation and maintenance procedures.

Conclusions:

Within the scope and constraints of the review performed, the Seabrook Station TS appear to accurately reflect the plant design and licensing basis, and regulatory requirements. While discrepancies were identified, the existing processes and controls appear to adequately communicate changes to the design and licensing bases, and seem effective in translating and maintaining the design bases requirements in the TS.

The assessment concluded that the types of discrepancies reflect the need for enhanced communications between those responsible for the design basis and those responsible for the Technical Specifications with respect to the purpose of the specifications. The requirement for an expanded bases section in conversion to the Improved Standard Technical Specifications should resolve this concern (see Section VI).

F. Request (f): *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 or reconstitution programs have been completed or are being conducted, provide a description of the review programs, including identification of the systems, structures, and components (SSCs), 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 SSCs and plant-level design attribute reviews, the expected completion date, and method of SSC prioritization used for the review.*

From construction through operation of Seabrook Station, North Atlantic has always recognized the importance of a fully integrated configuration management program that identifies existing plant design requirements; controls changes so that the plant structures, systems, components conform to those requirements; and ensures that the plant's physical and functional characteristics are accurately reflected in plant documents.

On July 8, 1987, North Atlantic assumed direct control of all site engineering and design activities from the Architect Engineer. All Seabrook Station engineering and design activities were accomplished in accordance with its own manuals and procedures from that date forward, although many of the documents and data bases were physically held by UE&C or in UE&C systems. To reach this point, we undertook a number of substantial design review efforts.

Design computer equipment/data lists such as the IE Equipment List, Electrical Bill of Materials List, Standard Instrument Schedule, Master Equipment List, Valve List, Line List, and Cable Schedule Program had been turned over to the utility prior to July 8, 1987. Computerized design document change tracking systems that were also turned over allowed the company to know and maintain the status of those documents. From 1986 through 1991, North Atlantic retrieved approximately 200,000 documents through an aggressive turnover process with the architect

engineer. Included in the turnover were approximately 50,000 drawings, 80,000 vendor drawings, 40,000 calculations, 1,000 specifications, 1,500 procedures, 1,000 security documents, and 2,000 equipment qualification records.

During this time frame, North Atlantic also began two important permanent programs, the Engineering Design Standards Program and the Design Basis Document Program. These were part of a continuing effort to expand on the philosophy of having a fully integrated and utility controlled configuration management program.

Engineering Design Standards

The Engineering Design Standard Program was started in 1986 in order to maintain consistency and continuity with the development of the original design of Seabrook Station by capturing existing Architect Engineer technical practices and methodology used in the design development of Seabrook Station. The Engineering Design Standards provide in-depth engineering knowledge in specific topical areas such as Seismic Qualification and Electrical Separation criteria.

Utility industry perspectives and information are assessed to determine the best practices that can be incorporated into the Engineering Design Standards Program. These are then used to enhance the standards and improve their ability to support the day to day operation of the plant. Engineering Design Standards not only support operations but improve the overall implementation of the design control and configuration management programs. To date, 25 Engineering Design Standards have been completed. They are listed in Table 4.

Design Basis Documents

A Design Basis Document (DBD) Program was started in 1988. The DBD Program consists of DBDs and System Description Documents (SDDs).

DBDs are being developed on:

- safety related systems and programs,
- important to safety systems and programs, and
- systems and programs that may have a potential impact on safety related systems and programs.

SDDs are developed on non-safety related systems. SDDs describe the process conditions that the system will be subjected to and include the current drawings, specifications and interface information.

DBDs/SDDs include current drawings, specifications and interface information as well as a summary of the design bases with source references and rationales. Functional requirements, regulatory commitments incorporated into plant design, owner preferences and plant form, fit and function changes are maintained current in the DBD.

DBDs and SDDs are not intended to supersede existing specifications, UFSARs or other requirements. The intent of a DBD/SDD is:

- to assemble program, system and component design basis information in one concise document for easy retrieval and update,
- to aid in making decisions on plant modifications,
- to provide background for evaluating technical specification issues,
- to provide information for making 10 CFR 50.59 Safety Evaluations, and
- to aid plant staff in support of maintenance and operation functions.

Seabrook Station has utilized the Nuclear Management and Resources Council (NUMARC) Report 90-12, Design Program Guidelines, issued October 1990 in the creation and formulation of DBDs. An engineering instruction provides the methodology for their preparation and control. It specifically provides guidance and requirements to:

- Establish program or system boundaries—The P&IDs, Schematics, Loops and Logics, or one-lines are used to establish fluid system boundaries, and the associated electrical systems, instrumentation and control boundaries for each DBD.
- Locate source documents—Project documentation associated with the program or system which establishes the design basis are collected.
- Prepare the DBD—Each DBD is prepared and processed for interdisciplinary review. Comments are resolved and incorporated into the DBD for submittal for final management review and approval.

Once approved, the DBD is distributed as a controlled document through the Records Management System to appropriate Seabrook organizations. Thereafter, each DBD is evaluated for updating purposes on a periodic basis. DBDs are reviewed when preparing design changes to ensure that the design bases are met or revised as appropriate. To date, 33 DBDs (Table 5) and 5 SDDs (Table 6) have been completed.

North Atlantic intends to continue to develop DBDs. To date, we have the completed about five DBDs and/or SDDs per year. The pace and priority of future development will depend on a number of factors. One such factor is our intention to perform an UFSAR review (see Section VI). Although we are still evaluating the scope of this review, it will use many of the same engineers who would be drafting the DBDs and SDDs. In establishing the future order of the DBD/SDD development, it is our intent to place a priority on the Maintenance Rule Systems, which are both Safety Related *and* Risk Significant. Maintenance Rule Systems, which are either Safety Related *or* Risk Significant will follow. Operating experience, regulatory issues and the demonstrated track record for reliability of systems and components both at Seabrook and in the industry at large will also be important inputs for determining the DBD program priorities. Because these factors change from time to time, we recognize the DBD Program as a continuous or 'living' process that will last for the life of the unit. The DBD Program continues

to be an important aspect of fully implementing a complete and comprehensive Configuration Management Program.

VI. ONGOING AND FUTURE ACTIONS

There are a number of activities that were started prior to, and independent of, the receipt of the NRC's letter of October 9, 1996 which address some of the concerns expressed in that letter. These are summarized below under **Ongoing Actions**. North Atlantic is also making a number of commitments as a result of the reviews and assessments conducted in support of this response to the NRC's letter. These commitments, listed under **Future Actions**, are designed to ensure that the weaknesses already identified are addressed and that any other insights on areas that warrant increased attention and improvement are extracted from the data generated by the effort.

A. Ongoing Actions

Procedure Upgrades—In 1994, North Atlantic began a major effort to review, consolidate, re-format, simplify and clarify about 3000 procedures over about a five year period. As discussed in the Response to Request (b), the methods and priorities of the upgrade program are currently being re-evaluated. The program, however, is going forward and will be implemented as close to the original schedule as reasonable.

Corrective Action Program Improvements— The response to Request (d) describes some of the improvements that have been, and continue to be, made in the Corrective Action Program. These include a gradual transition to the use of the ACR as the single adverse condition reporting document and a significant lowering of the threshold for initiating an ACR. This resulted in a dramatic increase in the number of ACRS written in 1996. We believe this is a positive sign. The Station Director has emphasized that if there is a question as to whether a condition is adverse, it is important to report it rather than "sit" on it. Enhancements to the corrective action program will continue.

Human Error Reduction and Root Cause Training—Seabrook Station has retained Performance Improvement International (formerly FPI) to conduct Human Error Prevention Training for managers, supervisors and individual employees. There is a two-day course for managers and supervisors and a one-day course for other employees. The purpose of the Root Cause Training

is to provide investigators with the tools to analyze human errors and organizational and programmatic failures, understand distribution of the causes of the failures in the industry, and recommend proven corrective actions based on the root causes/failure modes identified. The expected result is an improvement in the quality of root cause analyses and resulting improvement in the prevention of recurring events. North Atlantic has also contracted with Performance Improvement International for Technology Transfer Training so that we will have qualified, certified instructors for human error prevention training on staff.

UFSAR Maintenance and 10 CFR 50.59 Evaluation Improvements—In addition to lowering the ACR threshold, North Atlantic management has been emphasizing the need to critically review the accuracy of the UFSAR at every opportunity. This resulted in an increase in the number of ACRs related to UFSAR discrepancies in 1996. A group of these ACRs were subjected to a detailed common cause analysis to determine common conditions, problems and generic causes of the inconsistencies. As a result of this analysis, a number of recommendations have been approved. They include:

- Development of more guidance and clearer criteria for 10 CFR 50.59 screenings and safety evaluations.
- The requirement that a written basis be documented to support answers to the 10 CFR 50.59 screening questions for procedure changes as has been done for design changes.
- Making available LAN-based software to efficiently conduct searches to determine impacts of changes on the UFSAR.
- Additional training in the 10 CFR 50.59 and UFSAR update processes as well as in the lessons learned from the UFSAR common cause analysis.
- Provide follow-up surveillance and oversight to ensure that the corrective actions applied to prevent recurrence of UFSAR deviations are working.

These approved recommendations are currently being scheduled for implementation.

Design Basis Document Program—As discussed in the Response to Request (f), Seabrook Station has had a DBD program in place since 1988. To date, 33 DBDs have been developed. The future DBD development will use the Maintenance Rule criteria to help establish priorities.

Standard Technical Specifications Program—North Atlantic is in the process of converting to Standard Technical Specifications. As part of this effort, additional Technical Specification verifications will be conducted. Standard Technical Specification submittals are expected to complete by about July, 1998.

B. Future Actions

Dispositioning of Findings

- Tracking, Evaluation and Dispositioning of Findings—All ACRs generated by the reviews and assessments will be evaluated and dispositioned in accordance with the corrective action process. Findings that did not meet the threshold criteria for an ACR and recommendations from the review and assessment team will be tracked either on an Engineering Work Request, a procedure change recommendation or by being inputted directly into the Station's action item tracking system. They will all be evaluated and properly dispositioned.

- ACR Common Cause Analysis—North Atlantic will contract with Performance Improvement International to perform a common cause analysis of a population of ACRs that will include all of the appropriate ACRs generated by the self-assessment. This analysis will identify any additional areas that warrant increased attention, determine the causes of the discrepancies and identify corrective actions that will prevent recurrence.

UFSAR

- UFSAR Common Cause Analysis—To address the UFSAR discrepancies, North Atlantic will conduct an UFSAR common cause analysis that will include the UFSAR-related discrepancies found during the vertical slice and other reviews conducted for this report. Once the causes of these discrepancies are identified, we will consider the actions already being taken to improve UFSAR accuracy and determine what additional actions may be needed.

- UFSAR Comprehensive Review—North Atlantic will conduct a comprehensive UFSAR review. Its scope and methods may be impacted by the findings from the common cause analysis. Details on this review will be documented in a separate submittal to the NRC.

Calculation Common Cause Analysis—Calculation-related discrepancies from all of the reviews and assessments will be subjected to a common cause analysis to identify any common conditions, problems and generic causes. An action plan will then be developed to address identified weaknesses.

Engineering Topical Areas—Plans will be developed to provide greater structure, visibility and organizational understanding for the Engineering Topical Areas that were reviewed during the self-assessment (see Response to Request (e)).

Table 1

LIST OF TOPICS FOR CATEGORIZING FINDINGS

1.	Design Bases	1. Calculations and Analysis 2. Drawings 3. Specifications 4. Design Bases Documents / System Descriptions 5. Equipment Lists 6. Engineering Procedures 7. Change Documents
2.	Licensing Documentation	8. UFSAR 9. Tech Specs/Tech Requirements Manual 10. License Commitments 11. GLs, INs, IEBs
3	Operational Philosophy	
	a) Operations	12. Operations Procedures 13. Temporary Modifications 14. Operator Workarounds
	b) Maintenance	15. Maintenance Procedures 16. Vendor Information 17. Preventive Maintenance
	c) Surveillance	18. Surv Tests and Inspect. 19 Sect. XI, Pump & Valve Tests
4.	Physical Plant	20. As-Built Verification 21. Specialty Item Walkdowns 22. Ops Critical Drawings
5.	Programs Processes & Procedures	23. Corrective Action 24. Audits & Assessment 25. Interviews 26. Qual. Records (Training) 27. Mature Work Plan (Backlog)

Table 2

Scope of Vertical Slice System Reviews

System	Description	Scope
125 VDC System	The Station 125 VDC Power System is comprised of the Station batteries, battery chargers, and dc distribution equipment. It provides power for direct current load groups, vital control and instrumentation systems, and control and operation of Class 1E and non-Class 1E electrical equipment.	The areas selected for review included: • The bases, adequacy, documentation, physical condition, etc. of: Battery rating Battery charging rate Switchgear rating Distribution panels • Electrical Coordination of the 125 VDC System • Short circuit current calculations and design bases • Temp mods, Maintenance, surveillance procedures • Electrical separation • Seismic equipment qualification
Emergency Diesel Generator System	The functions of the Emergency Diesel Generator System are to provide an on-site source of emergency electrical power in the event of a plant loss of coolant accident (LOCA) or to provide power in the case of a loss of off-site power (LOOP) event. The review focused on the mechanical, instrumentation and control, and protection portions of the emergency diesel generator sets due to the fact that the	The areas selected for review included: • Bases, supporting calculations, documentation, testing, etc. for: EDG rating Standby conditions and starting Loading and sequencing Fuel consumption Voltage and frequency regulation • Calculations, specifications and modifications since startup for: Building ventilation Engine cooling Start air system Lubrication system

Table 2 (continued)

Scope of Vertical Slice System Reviews

System	Description	Scope
	electrical portion of the system has been examined by the NRC in an in-depth EDFSI in April and May of 1993.	Intake and Exhaust system • Compliance of EDG control and protection system with industry codes • Walkdown of the system • Review of procedures to ensure commitments have been met
Primary Component Cooling Water System	The function of the PCCW system is to remove heat from several components during both normal operation and under accident conditions and to transfer this heat to the Service Water system. The PCCW system consists of two independent and redundant loops (A and B).	The areas selected for review included: • Heat removal requirements and calculations for the PCCW, RHR and CBS heat exchanger • Design and physical configuration, license commitments, surveillance and testing, maintenance procedures and modifications for: PCCW, RHR and CBS heat exchangers PCCW temperature control valves PCCW head tank PCCW pumps and valves RHR heat exchanger isolation valve • Electrical separation and EQ • Piping pressure boundary integrity and structural • Implementation of the recommendations from the recent SSFA of PCCW
Emergency Feedwater System	The functions of the Emergency Feedwater System are to provide the capability to remove heat from the Reactor Coolant System during emergency conditions when the Main Feedwater System is not available, including small	• The areas selected for review included: • Turbine driven feed pump • Motor driven feed pump • Startup feedwater pump • CST and alternate water supply • Inservice test program for pumps and valves

Table 2 (continued)

Scope of Vertical Slice System Reviews

System	Description	Scope
	break LOCA cases. The EFW System steam driven pump will operate during a Station Blackout. The EFW System also provides a flow path for water to the steam generators during startup feedwater pump operation.	• Initiation and control systems • Capabilities during Station blackout • Containment isolation • Diversity and single failure, walkdowns and interviews • Electrical separation • Piping Design, HELB
Emergency AC System	The function of the Emergency AC System is to provide ac power for the safety systems to accomplish their safety functions.	The areas selected for review included: • Any site or service conditions that limit capability of system • Capacity capabilities of the system • Actual load imposed on equipment. • Actual ratings of equipment • Automatic and manual controls consistent with UFSAR and Licensing commitments • Protection and protective actions are consistent with UFSAR • Equipment voltage limits and ratings • Equipment fault current ratings • Safety classifications, 10CFR50.59 safety evaluations • service conditions, redundancy and electrical separation
Residual Heat Removal System	The Residual Heat Removal System has many functions, but the most significant safety functions are to provide low pressure safety injection and cooling to the RCS after a postulated Loss Of Coolant Accident (LOCA) and backup	The areas selected for review included: • Design features to ensure proper system alignment • Provisions to prevent suction line overpressure • Supporting documentation and implementation of setpoints • Documented commitments for accident monitoring instrumentation • Environmental qualification and flood level considerations • Temp mods, walkdowns, corrective action, audits and assessments,

Table 2 (continued)

Scope of Vertical Slice System Reviews

System	Description	Scope
	low temperature overpressure protection in conjunction with the PORVs. These were the focus of this review along with system alignment.	• Piping and pump design including material selection, safety class
Plant Protection System	The Plant Protection System is comprised of portions of several plant systems whose functions are to monitor critical plant parameters and initiate a reactor trip and actuate any appropriate Engineered Safety Features (ESF), if safety limits are exceeded.	Areas selected for review included: • Sensing and signal processing for containment pressure and steam generator level. • Logic and actuation relays for containment pressure and steam generator level loops. • Testing and maintenance records and vendor recommendations for the Reactor trip breakers • Documentation, control diagrams and surveillance procedures for the ESF actuation devices • Manual controls, indicators and alarms associated with the PPS • PPS setpoints • Surveillance Testing • Licensing commitments, results of audits and assessments • Independence, instrument tubing

Table 3

Scope of the Engineering Technicai Program Reviews

Program	Description	Scope
Environmental Qualification (EQ)	The EQ Program provides the controls for the qualification of all important safety electrical equipment. These licensing and design requirements are specified in 10 CFR 50.49.	The areas selected for review included: • The licensing basis of the EQ Program which includes regulatory correspondence which establishes conformance to 10 CFR 50.49 • The EQ Program design documents developed to meet the requirements of 10 CFR 50.49 • Implementation of EQ Program requirements such as replacement intervals and EQ required maintenance • Operational procedures used to operate and maintain EQ Program equipment to insure consistency with the licensing basis • Component walkdowns to ensure that the design basis is adequately reflected in the physical plant configuration
Fire Protection (Appendix R)	Programmatic requirements for the safe shutdown capability from fire events are maintained within this program. The requirements are defined in 10 CFR 50 Appendix R.	The areas selected for review included: • Design bases and licensing documentation • Operations, testing and maintenance procedures • Programmatic controls for Appendix R Safe Shutdown Compliance • Functional requirements; system, component, and cable selection; process monitoring; emergency lighting; and manual action procedures (following the format of NRC Inspection Procedure 64100) • Commitments, deviations, and evaluations • Program assumptions in the context of current expectations from recent NRC inspections of other utilities • Long term configuration control to ensure continued compliance
Station Blackout (SBO)	The programmatic requirements for the evaluation of the ability to cope with a loss of all AC power (station blackout) are documented	The areas selected for review included: • Implementation of the SBO analysis and verification that SER commitments were implemented and closed

Table 3 (continued)

Scope of the Engineering Technical Program Reviews

Program	Description	Scope
	and maintained within this program. The requirements to withstand a loss of both offsite power and onsite power are specified in 10 CFR 50.63.	• Proper consideration of SBO in the development and revision of related documentation (UFSAR, design changes, procedures, etc.) • Validity of the SBO assessments relative to current revisions of referenced documents • Licensing commitments related to the SBO Safety Evaluation • Five initiatives defined in NUMARC 87-00 • Review of several procedures that could affect SBO • Emergency SBO lighting guidelines of GL 81-04
Inservice Testing (including Appendix J)	The inservice testing (IST) program addresses the testing of ASME Class 1, 2 and 3 pumps and valves to assess their operational readiness. The Appendix J testing program addresses containment leak rate testing. The programmatic requirements for IST and Appendix J Testing are specified in 10 CFR 50.55a and 10 CFR 50 Appendix J.	The areas selected for review included: • The Seabrook Inservice Testing Reference Manual (SITR), which covers the first ten year interval program plan for IST of pumps and valves • The specific Station engineering and operating procedures that implement the IST program mechanics and collection of test data • The Seabrook Leakage Test Reference Manual (SLTR) for the Primary Reactor Containment Leakage / Test Program as required by 10CFR50 Appendix J and Seabrook Station Technical Specifications • The test results of three Type A tests conducted from 1986 to date, and the Type B and C tests
ASME XI (ISI)	The Inservice Inspection (ISI) Program provides the controls and methodology to test, evaluate and document all piping code repairs. These licensing and design requirements are specified in Regulatory Guide 1.147 and ASME Section XI.	The areas selected for review included: • UFSAR, Technical Specifications, SER, Generic Letters, NRC correspondence and inspections as well as Seabrook control and implementing documents were reviewed against the Seabrook ISI Program as defined and controlled by the SIIR • ASME Section XI requirements regarding performance of nondestructive examinations of Class 1, 2, 3 components and their supports.

Table 3 (continued)

Scope of the Engineering Technical Program Reviews

Program	Description	Scope
		• ASME Section XI requirements concerning repairs and replacements. • Augmented ISI. • ISI requirements for steam generators tubes.
Electrical Separation	This program provides the requirements for independence and separation of electrical circuits and equipment. The specific programmatic requirements are defined in Regulatory Guide 1.75 Revision 2, IEEE 384-1975 and 10 CFR 50 Appendix R.	The areas selected for review included: • The implementation and closure of inspection findings and commitments from the NRC Electrical Distribution Functional Inspection (EDSFI) of 1993 and follow-up inspection of September 1994 • Seabrook's response to unresolved items identified during the EDSFI Inspection • NRC Inspection Reports for issues relating to Electrical Separation and Associated Circuits • Procedures and programs developed to close NRC unresolved issues • Interviews to verify effectiveness • Selected area walkdowns to verify that internal cabinet cable and external cable tray and raceway separation requirements are met
Post Accident Monitoring (RG 1.97)	This program provides the requirements for the accident monitoring instrumentation. The programmatic requirements are defined in Regulatory Guide 1.97 Revision 3.	The areas selected for review included: • Maintenance of the accident monitoring instrumentation (AMI) List, revising the EOPs, and storing the EOP change • Tech Spec 6.7.6.f Conformance, with requirements for Design Category 2 & 3 AMI • NRC Inspection Report from the R.G. 1.97 implementation audit, including the commitments and one unresolved issue • UFSAR Section 7.5 where commitments were made to include information on instrumentation in the operator training program
IPEEE (including	The IPEEE external events and hazards analysis addresses the	The areas selected for review included:

Table 3 (continued)

Scope of the Engineering Technical Program Reviews

Program	Description	Scope
external events)	effects of external events and hazards (i.e. tornado, earthquake, flood and snow). The specific requirements are defined in 10 CFR 50 Appendix A, General Design Criteria (GDC) 2 and 4 and Appendix A to 10 CFR Part 100.	• Guidance provided in the NRC Generic Letter and NUREG-1407 • The Seabrook IPEEE Report and associated correspondence, including a sample of six commitments or attributes • The licensing basis associated with External Events, including the applicable sections of the UFSAR, NRC SER and other documents, including a sample of ten commitments or attributes
High Energy Line Break (HELB)	This program provides the requirements for postulated piping failures in high. The specific requirements are defined in NUREG-0800 and Standard Review Plan SRP 3.6.2.	The areas selected for review included: • UFSAR, the NRC SER and amendments and the SRP to assess the licensing basis and commitments • UFSAR, Chapter 3.6 to ensure it reflects the licensing requirements provided in SRP Sections 3.6.1 and 3.6.2 • Associated NRC correspondence was reviewed to ensure that the licensing basis is current • Programmatic interface between the HELB Program and affected engineering standards • Selected plant design changes to ensure that HELB requirements are consistently addressed • Field walkdowns of design changes and selected HELB related plant features

TABLE 4

COMPLETED ENGINEERING DESIGN STANDARDS (EDS)

NUMBER	TITLE	REV	DATE
36120	Structural Steel	1	08/30/96
36140	Concrete Modifications	1	04/11/96
36160	Piping Systems	2	02/16/96
36260	Pipe Supports	1	01/13/95
36300	HVAC Ducts and Supports	0	08/04/87
36340	Conduit Supports	0	08/04/87
36360	Class IE Equipment Seismic Qualification	0	08/04/87
36380	ASME Equipment Seismic Qualification	0	08/04/87
36400	Seismic Qualification of Miscellaneous Safety-Related and Non-Safety Related Equipment Located In Safety-Related Buildings	0	08/04/87
36420	Separation Criteria	0	04/07/88
36460	Structural Evaluation of Erosion Corrosion Thinning in Carbon Steel Pipe	0	09/23/92
37120	HELB Program Design Bases and Implementation	1	12/01/96
37160	Line List	0	04/19/88

Table 4 (continued)

COMPLETED ENGINEERING DESIGN STANDARDS (EDS)

NUMBER	TITLE	REV	DATE
37165	Piping Bill of Material	0	03/15/93
37180	Classification of Structures, Systems and Components	3	11/03/95
37190	Locked Valve Criteria	2	07/22/91
37200	Classification of Additives to Plant Systems	0	08/14/90
37210	ALARA Design Considerations	1	01/21/93
37510	Valve Selection Guide for ASME III, Class 1, 2, 3, and ANSI B31.1 Pipe Installations	0	01/30/89
38120	Electrical Separation, Associated Circuits and Other Unique Criteria	3	04/27/95
38160	Fire Protection for Safe Shutdown	0	08/04/87
38200	Revision of Electrical Bus Failure Analysis Report	0	06/02/88
39140	Instrument Setpoint	1	10/25/93
39160	Main Control Boards Modification	1	04/11/96
39180	Review and Maintenance of IE Equip List	0	08/04/87

TABLE 5

COMPLETED DESIGN BASIS DOCUMENTS

Number	Title	Revision	Effective Date
DBD-CBA-01	Control Room Complex HVAC Systems	0	03/24/94
DBD-CBA-02	Control Building Electrical Areas Heating and Ventilating Systems	0	12/15/94
DBD-CC-01	Primary Component Cooling Water System	1	01/04/95
DBD-CO-01	Condensate System	0	07/20/94
DBD-DAH-01	Diesel Generator Building Heating and Ventilating Systems	0	02/02/94
DBD-DG-01	Emergency Diesel Generator Mechanical	0	01/27/95
DBD-EAH-01	Containment Enclosure Cooling and Exhaust Filter Systems	0	12/22/94
DBD-ED-01	Technical Requirement 13 Containment Penetration Conductor Overcurrent Protective Devices	3	05/14/96
DBD-ED-02	Technical Requirement 14 Motor-Operated Valves with Thermal Overload Protection Devices Operable at all Times	1	05/14/96
DBD-ED-03	Technical Requirement 15 Protective Devices for Non-1E Circuits Connected to Class 1E Power Sources	1	05/14/96
DBD-ED-04	120 VAC Vital & Non-Vital Instrument Power Systems	0	05/14/96
DBD-ED-05	125 VDC System	0	05/14/96
DBD-EFW-01	Emergency Feedwater System	0	07/26/89

TABLE 5

COMPLETED DESIGN BASIS DOCUMENTS
(Continued)

Number	Title	Revision	Effective Date
DBD-EQ-01	Environmental Qualification of Electrical Equipment Important to Safety	0	12/10/93
DBD-ESF-01	Engineered Safety Features Response Times	0	09/26/96
DBD-FAH-01	Fuel Storage Building Heating and Ventilation Systems	0	12/09/93
DBD-FP-01	Appendix 'R' Emergency Lighting	1	11/15/96
DBD-FP-02	Fire Detection Systems	0	01/23/90
DBD-FP-03	Fire Suppression Systems	0	05/04/90
DBD-FP-04	Fire Rated Walls, Floors, and Ceiling Assemblies	0	07/19/90
DBD-FP-05	Fire Hydrants, Hose Stations & Miscellaneous	0	10/10/90
DBD-FP-06	Fire Rated Doors, Dampers, Conduit Wrap and Heat Shields	0	01/14/91
DBD-FP-07	Fire Rated Penetration Seals	1	08/16/96
DBD-FW-01	Feedwater System	0	03/14/93
DBD-MET-1	Meteorological Monitoring System	1	06/11/96
DBD-NI-01	Nuclear Instrumentation System	1	09/16/96
DBD-PB-01	Plant Barriers	0	10/23/92
DBD-RC-01	Reactor Coolant System	0	06/11/96
DBD-RC-02	Setpoint Methodology for Protection Sys	0	08/31/92

TABLE 5

COMPLETED DESIGN BASIS DOCUMENTS
(Continued)

Number	Title	Revision	Effective Date
DBD-RH-01	Residual Heat Removal System	0	10/22/93
DBD-RM-01	Radiation Monitoring	1	08/16/96
DBD-SW-01	Service Water System	1	09/16/96
DBD-VB-01	Loose Parts and Valve Flow Monitoring Systems	1	06/11/96

Table 6

COMPLETED SYSTEM DESCRIPTION DOCUMENTS

Number	Title	Revision	Effective Date
SDD-CWS-01	Circulating Water System	0	06/14/93
SDD-HT-01	Heat Tracing System	0	12/30/94
SDD-RMW-01	Reactor Makeup Water	0	05/11/93
SDD-SA-01	Service (Compressed Air System)	0	08/23/95
SDD-VG-01	Equipment Vent Gas	0	12/09/93

Figure 1

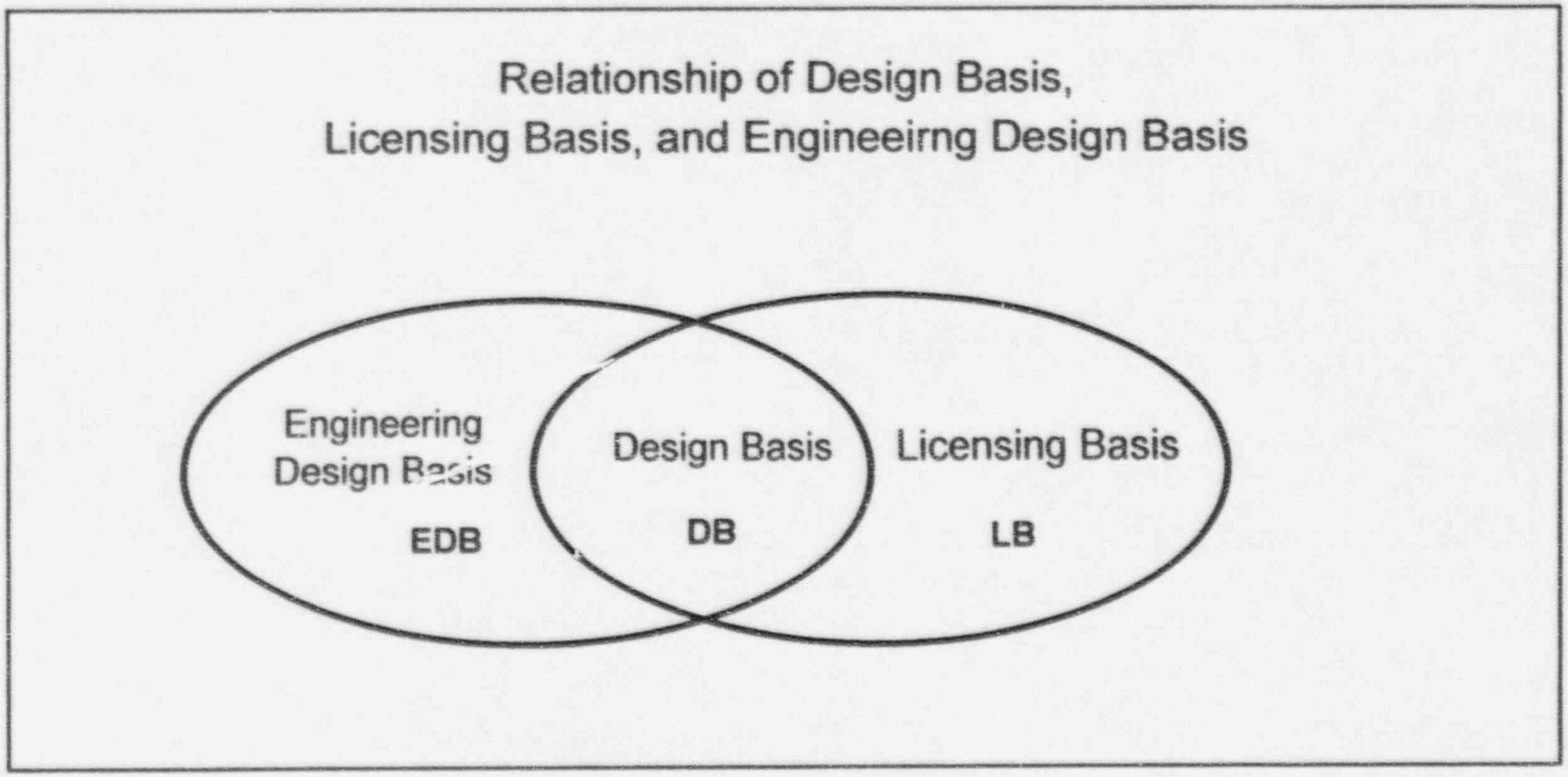

Relationship of Design Basis,
Licensing Basis, and Engineeirng Design Basis
Engineering
Design Basis
EDB
Design Basis
DB
Licensing Basis
LB

Figure 2

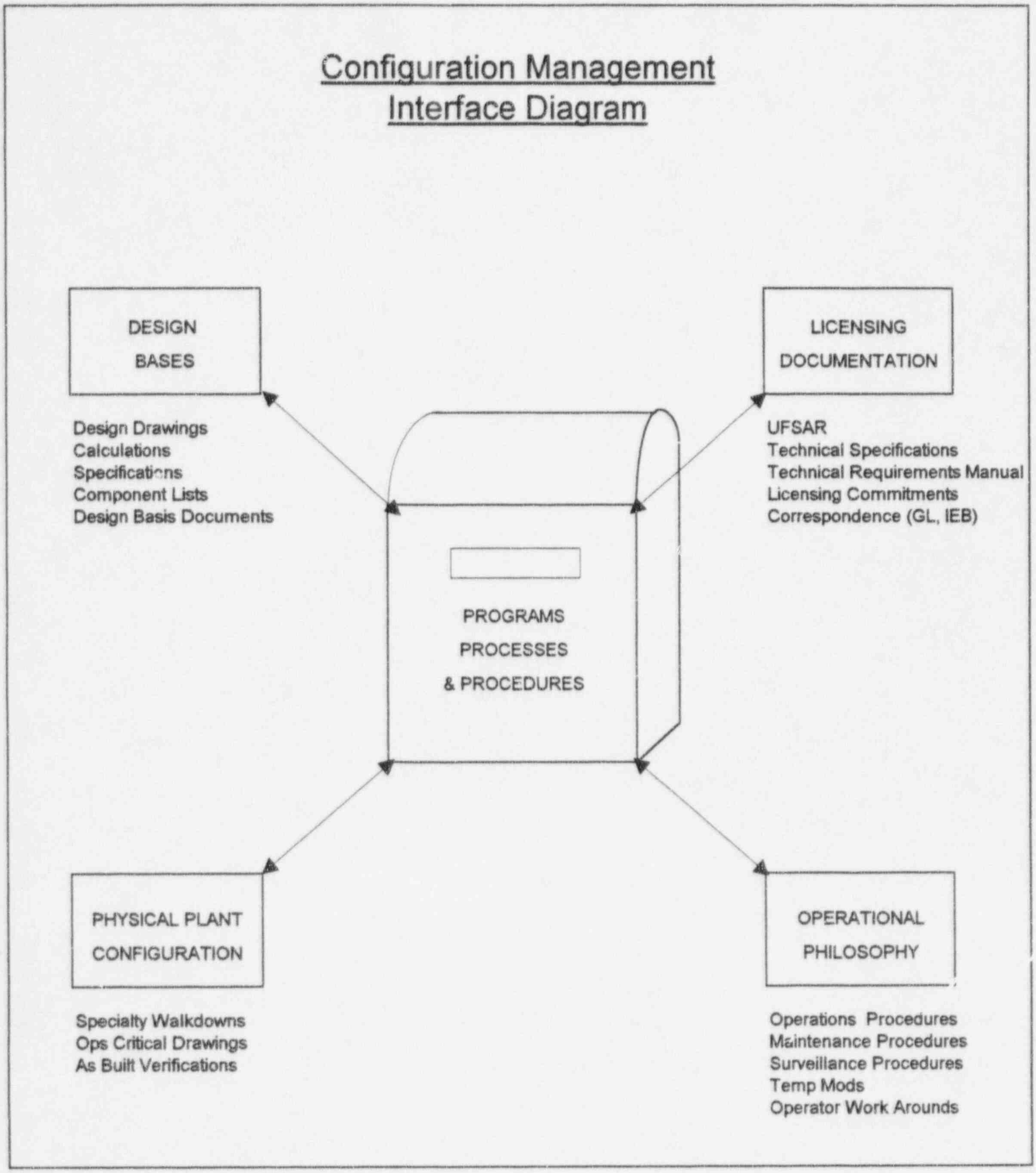

Configuration Management
Interface Diagram
DESIGN
BASES
Design Drawings
Calculations
Specifications
Component Lists
Design Basis Documents
LICENSING
DOCUMENTATION
UFSAR
Technical Specifications
Technical Requirements Manual
Licensing Commitments
Correspondence (GL, IEB)
PROGRAMS
PROCESSES
& PROCEDURES
PHYSICAL PLANT
CONFIGURATION
Specialty Walkdowns
Ops Critical Drawings
As Built Verifications
OPERATIONAL
PHILOSOPHY
Operations Procedures
Maintenance Procedures
Surveillance Procedures
Temp Mods
Operator Work Arounds

Figure 3

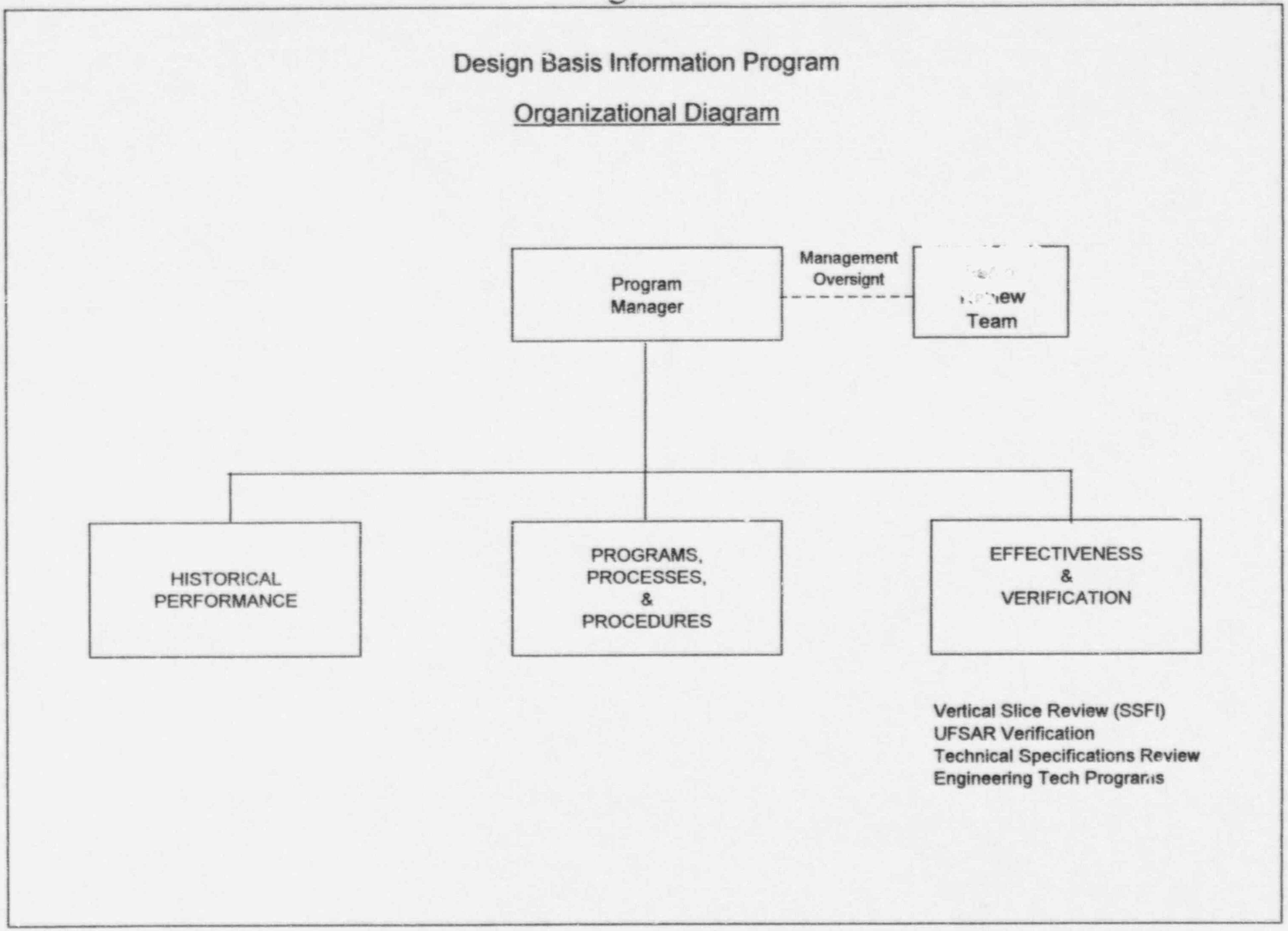
INITIAL SCOPE DEFINITION
SYSTEM VERTICAL SLICE REVIEW MATRIX

EMERGENCY FEEDWATER SYSTEM

EXAMPLE

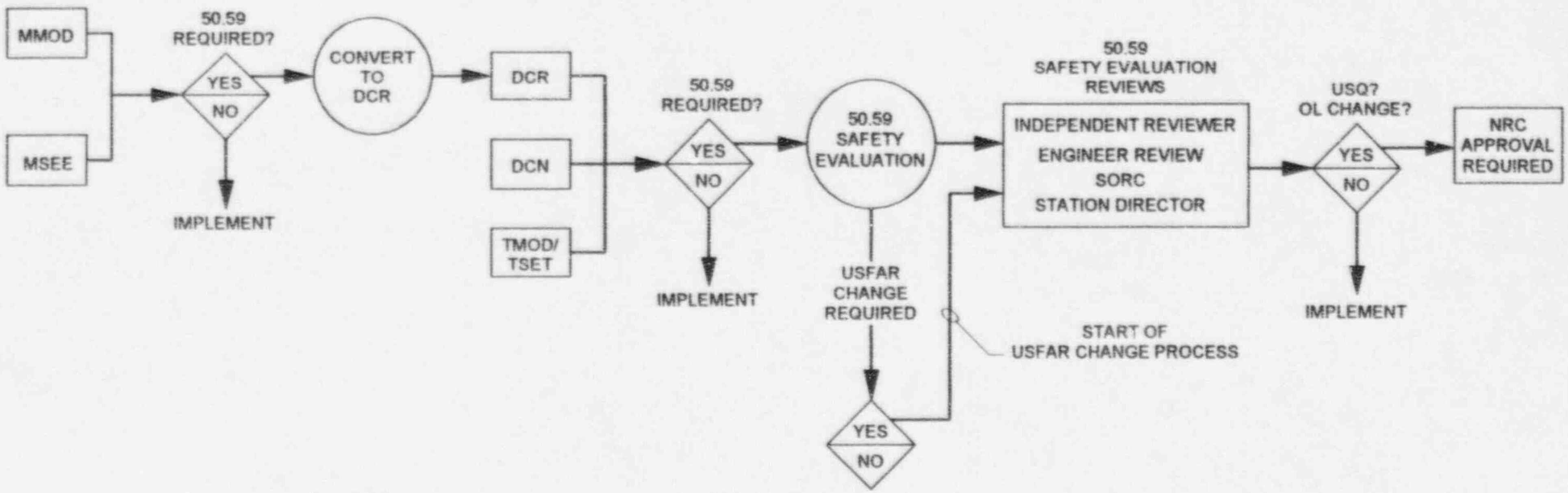
	TOPICS	Design Bases							Licensing Documentation				Ops			Maintenance			Surveillance		Physical Plant			Prog., Proces. & Proced.				
		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27
	ASSESSMENT ATTRIBUTES	Calcs/Analysis	Drawings	Specifications	DBD's & SD's	Equipment Lists	Engineering Procedures	Change Docs	UFSAR	Tech Specs / TRM	Licensing Commitments	Notices & Bulletins	Procedures	Temp Mods	Operator Work Arounds	Maintenance Procedures	Vendor Info	Preventive Maintenance	Surveillance Test & Inspections	Section XI IST/ Pump & Valve Tests	As-Built Verification	Specialty Item Walkdowns	OPS Critical Drawings	Corrective Action	Audits & Assessments	Interviews	Qual Records (Training)	Mature Work Plan (Backlog)
NO.																												
EFW01	TURBINE DRIVEN PUMP (1-FW-P-37A)	X	X	X	X			X	X	X	X	X				X	X	X	X		X							
EFW02	MOTOR DRIVEN PUMP (1-FW-P-37B)	X	X	X	X			X	X	X	X	X				X	X	X	X		X							
EFW03	STARTUP FEEDWATER PUMP (1-FW-P-113)	X	X	X	X			X	X	X	X	X				X	X	X	X		X							
EFW04	CST & ALTERNATE WATER SUPPLY	X	X		X				X	X	X					X			X		X							
EFW05	PUMP & VALVE TESTING	X		X	X				X	X	X	X							X									
EFW06	INITIATION & CONTROL	X	X		X			X	X	X	X	X							X	X	X							
EFW07	STATION BLACKOUT OPERATION	X			X			X	X		X	X	X						X									
EFW08	CONTAINMENT ISOLATION		X	X	X			X	X	X	X	X				X	X	X	X	X	X							
EFW09	EFW OVERALL													X	X					X		X	X	X	X	X	X	X
EFW10	ELECTRICAL SEPARATION	X	X	X	X			X	X	X		X										X	X		X			X
EFW11	ME/HE LINE BREAKS	X	X	X			X	X	X																			
EFW12	PIPING DESIGN	X	X	X			X	X	X													X	X					
EFW13	SEISMIC QUALIFICATION	X					X	X	X													X						
EFW14	EQUIPMENT QUALIFICATION		X	X		X										X	X	X	X	X			X					

Figure 4



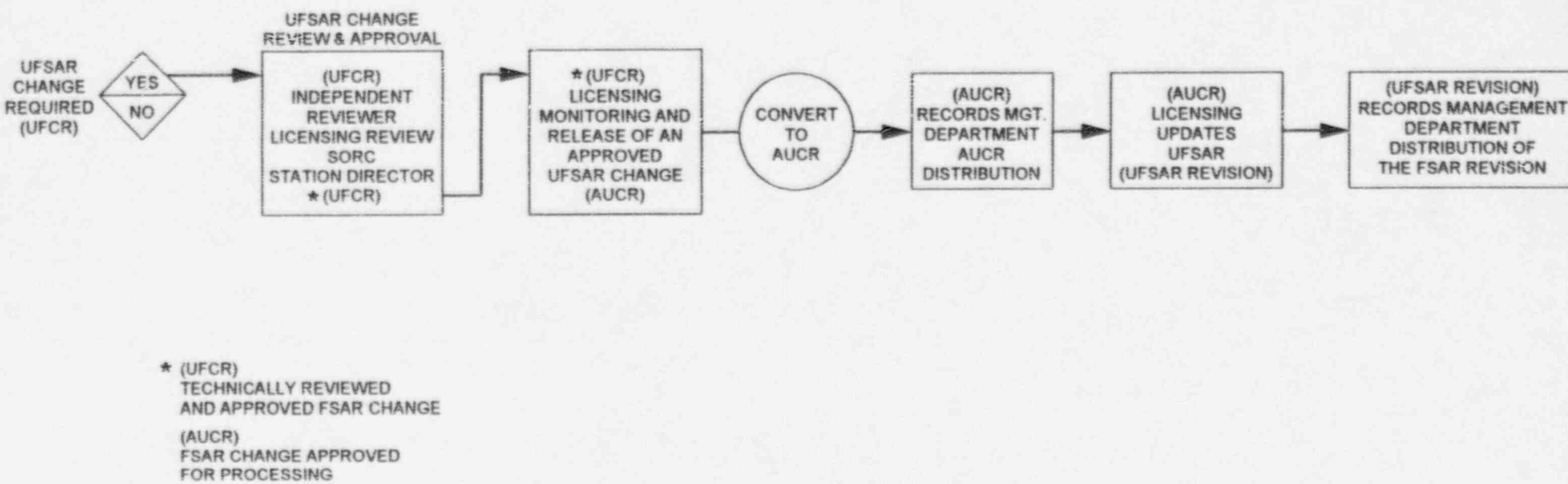
Design Basis Information Program
Organizational Diagram
Program Manager
Management Oversight
Team
HISTORICAL PERFORMANCE
PROGRAMS, PROCESSES, & PROCEDURES
EFFECTIVENESS & VERIFICATION
Vertical Slice Review (SSFI)
UFSAR Verification
Technical Specifications Review
Engineering Tech Programs

Figure 5



DESIGN CHANGE SAFETY EVALUATION PROCESS

Figure 6



UFSAR CHANGE PROCESS