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CP&L

Carolina Power & Light Company
PO Box 1551
Raleigh NC 27602

William S. Orser
Executive Vice President
Energy Supply

Serial: PE&RAS-97-006
February 11, 1997

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324 / LICENSE NOS. DPR-71 AND DPR-62

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400 / LICENSE NO. DPR-63

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / LICENSE NO. DPR-23

SUBJECT: RESPONSE TO REQUEST FOR INFORMATION PURSUANT TO 10CFR50.54(f) REGARDING ADEQUACY AND AVAILABILITY OF DESIGN BASES INFORMATION

Dear Sir or Madam:

This letter responds to the Nuclear Regulatory Commission (NRC) letter dated October 9, 1996, from Mr. James M. Taylor to Mr. Sherwood H. Smith, Jr., regarding the adequacy and availability of design bases information at Carolina Power & Light's (CP&L) nuclear plants. The letter was received by Mr. Smith on October 15, 1996. Therefore, CP&L is providing its response on or before February 12, 1997. I am responding on behalf of CP&L Chief Executive Officer, Mr. William Cavanaugh III, in my capacity as CP&L Executive Vice President - Energy Supply and Chief Nuclear Officer.

We acknowledge and fully accept our responsibility to operate and maintain our nuclear plants in accordance with the operating licenses and in a manner that protects the health and safety of the public. In meeting this responsibility, we understand the importance of operating and maintaining our nuclear plants consistent with design bases requirements.

We have expended considerable efforts to collect, compile, and evaluate data to respond to the NRC request. Collectively, the preparation and review of this response represents the collaborative efforts of a team of experienced and technically diverse individuals from our plants and corporate staff.

A074/1

This response is applicable to the Brunswick, Robinson, and Harris Nuclear Plants and consists of seven attachments. The first five attachments provide information requested by items (a) through (e) in the NRC letter. A sixth attachment responds to the request for information concerning the design bases document compilation and reconstitution efforts at the CP&L nuclear facilities. The seventh attachment provides a summary of the specific CP&L commitments or future enhancements to our programs and processes.

The programs and processes described in this submittal are not static, and the description of these programs and processes represents a single point in time. Accordingly, we will continue our efforts to improve our programs and processes that address design bases and configuration management.

To provide additional assurance that the design bases have been appropriately incorporated into the design, operation, and maintenance of the plants, we intend to take additional actions including design bases related self-assessments, implementation of improved standard technical specifications and selected plant specific initiatives. These actions are discussed in the responses to the specific requests and are summarized in Attachment G.

Based upon our assessments and reviews, we believe that reasonable assurance exists to conclude that overall, CP&L plant design bases at the Brunswick, Robinson, and Harris Nuclear Plants have been adequately incorporated through processes and programs into the plant design and applicable procedures. We acknowledge that deficiencies have been identified in the past. However, when their number, type, safety significance, and resulting corrective actions are considered in the whole, these deficiencies do not change our conclusion about the design bases today. Furthermore, we expect that additional deficiencies will be identified in the future by our self-assessment programs. These deficiencies will be addressed via the Corrective Action Program. Through this persistent assessment process and the resulting corrective actions, including continuing training programs, we will strive to continue to properly maintain processes, programs, procedures, and configuration consistent with the design basis.

As required by 10CFR50.54(f), this response is provided under affirmation.

Sincerely,

W. S. Orser
Executive Vice President, Energy Supply
Chief Nuclear Officer

Attachments

W. S. Orser, having been first dully sworn, did depose and say that the information contained in this submittal is true and correct to the best of his information, knowledge and belief.

(Seal) Susie G. Bunn 2/11/97

My commission expires: 4/20/97

SUSIE G BUNN
NOTARY
PUBLIC
JOHNSTON COUNTY, NC

c: Mr. L. J. Callan, Executive Director for Operations
Mr. S. J. Collins, Director, USNRC Office of Nuclear Reactor Regulation
Mr. L. A. Reyes, Regional Administrator, Region II
Mr. J. B. Brady, USNRC Senior Resident Inspector - HNP, Unit 1
Mr. B. B. Desai, USNRC Senior Resident Inspector - HBRSEP, Unit 2
Chairman J. A. Sanford - North Carolina Utilities Commission
Mr. N. B. Le, USNRC Project Manager, HNP, Unit 1
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP, Unit 2
Mr. C. A. Patterson, USNRC Senior Resident Inspector - BSEP, Unit Nos. 1 and 2
Mr. D. C. Trimble, Jr., USNRC Project Manager, BSEP, Unit Nos. 1 and 2

CAROLINA POWER & LIGHT RESPONSE TO NRC INFORMATION REQUEST

DESIGN BASES AND CONFIGURATION CONTROL

NRC Letter dated October 9, 1996
Issued Under 10CFR50.54(f)

Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Docket Nos. 50-325 and 50-324

Shearon Harris Nuclear Plant, Unit No. 1
Docket No. 50-400

H. B. Robinson Steam Electric Plant, Unit No. 2
Docket No. 50-261

EXECUTIVE SUMMARY

The Nuclear Regulatory Commission (NRC), by letter dated October 9, 1996, has requested licensees to provide, pursuant to 10CFR50.54(f), information regarding the adequacy and availability of design bases information. This submittal is the response of Carolina Power & Light Company (CP&L) for the Brunswick Steam Electric Plant, Unit Nos. 1 and 2; the Shearon Harris Nuclear Power Plant, Unit No. 1; and the H. B. Robinson Steam Electric Plant, Unit No. 2. This submittal provides a description of the design bases processes and programs for CP&L's nuclear plants. Following is a restatement of the NRC request for information and a summary of CP&L's response.

(a) Description of engineering design and configuration control processes, including those that implement 10CFR50.59, 10CFR50.71(e), and Appendix B to 10CFRPart50;

The response describes a combination of plant-specific and corporate processes, programs, and procedures to administer design and configuration control activities and to maintain the design bases of the plants. In this response, CP&L provides a description of : 1) Design Documentation types, 2) Configuration Management, Design Control, and Design Change Processes, 3) Implementation of 10CFR50.59, 4) Implementation of 10CFR50.71(e), and 5) Implementation of 10CFR50, Appendix B.

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

The response describes the current programs and processes governing the translation of design bases requirements into operating, maintenance, and testing procedures. The plant-specific response contains five subsections: 1) Summary, 2) Procedure Development and Control, 3) Assessments and Reviews, 4) Improvement Initiatives, and 5) Conclusion. Generally, the rationales for concluding that design bases requirements are translated into operating, maintenance, and testing procedures are based upon: established processes for procedure development and control of design bases information, assessment programs, and the Corrective Action Program.

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

The response describes both the current processes for the control of system, structure and component configuration, and the current testing and monitoring programs established to demonstrate the performance of the systems, structures, and components. The plant-specific responses contain six subsections: 1) Summary, 2) System, structures and components (SSC) Configuration Control, 3) SSC Performance, 4) Assessments and Reviews, 5) Improvement Initiatives, and 6) Conclusion. Generally, the rationales for concluding that system, structure, and component configuration and performance are consistent with the design bases are based upon: established programs to maintain the design bases, established post-maintenance and post-modification testing programs, surveillance testing programs, assessment programs, and the Corrective Action Program.

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

The response describes a combination of plant-specific and corporate processes, programs, and procedures established for the identification and correction of problems, including actions to determine the extent of problems, actions to prevent recurrence, and actions for reporting to the NRC when appropriate. These programs consist of the Corrective Action Program, including trending, augmented by several additional programs: 1) Self-Assessment, 2) Independent Assessment, 3) Corrective Maintenance, 4) Document Change, 5) Drawing Change, 6) Operating Experience Assessment, and 7) Employee Concern.

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

In determining the overall effectiveness of the current processes and programs used to maintain the configurations of the plants consistent with the design bases, CP&L: 1) reviewed the information presented in Attachments A through D and F, 2) considered the use of selected performance indicators, and 3) reviewed provisions for management assessments.

Based upon our assessments and reviews, we believe that reasonable assurance exists to conclude that overall, CP&L plant design bases at the Brunswick, Robinson, and Harris Nuclear Plants have been adequately incorporated through processes and programs into the plant design and applicable procedures. We acknowledge that deficiencies have been identified in the past. However, when their number, type, safety significance, and resulting corrective actions are considered in the whole, these deficiencies do not change our conclusion about the design bases today. Furthermore, we expect that additional deficiencies will be identified in the future by our self-assessment programs. These deficiencies will be addressed via the Corrective Action Program. Through this persistent assessment process and the resulting corrective actions, including continuing training programs, we will strive to continue to properly maintain processes, programs, procedures, and configuration consistent with the design bases.

(f) *The NRC has requested a description of any design review or reconstitution programs conducted or being conducted by CP&L.*

The programs which were undertaken for the compilation of design bases information at the three CP&L nuclear plants are described in this response. These programs resulted in a set of Design Basis Documents (DBDs) for each plant. The response describes for the plants' DBD effort including: 1) a description of the design bases information review programs, 2) identification of structures, systems, and components included, 3) identification of included plant-level topics (e.g., seismic qualification, consideration of high-energy and moderate-energy line break), 4) a description of the process to provide correctness and accessibility of the DBD information, and 5) a description of DBD maintenance.

Attachment G provides a listing of the CP&L commitments and future enhancements which will add to our confidence in the overall effectiveness of these programs.

Introduction and Methodology

This response represents the collaborative efforts of a team of experienced and technically diverse individuals from our plants and corporate staff.

Information from the last ten years was generally considered in the development of this response. The time frame for the Harris Nuclear Plant (HNP), which obtained its operating license after Three Mile Island, begins just prior to full power operating license issuance in January 1987. For the Robinson Nuclear Plant (RNP) and the Brunswick Nuclear Plant (BNP), the time frame (January 1, 1987 to January 1, 1997) permitted a review from the time period when Safety System Functional Inspections (SSFIs) were performed at both sites. The time frame also includes the BNP 1992 dual unit outage, and subsequent Unit 1 and Unit 2 restarts in February 1994 and May 1993, respectively.

Carolina Power & Light Company's (CP&L) response to NRC Item (a) is presented as a corporate response in Attachment A. The following five topics are discussed:

1) Design Documentation;
2) Configuration Management, Design Control, and Design Change Process;
3) Implementation of 10CFR50.59;
4) Implementation of 10CFR50.71(e); and
5) Implementation of 10CFR50, Appendix B.

Site-specific responses are presented for NRC items (b) and (c). CP&L considers this appropriate due to the differences in licensing, design, and operating experience of each site. Attachment B discusses procedure development and control. Attachment C discusses system, structure, and component (SSC) configuration and performance. Both Attachments B and C present results of pertinent internal and external assessments and a discussion of improvement initiatives in each area. CP&L's overall rationales for NRC items (b) and (c) are presented in the respective conclusion sections for Attachments B and C.

Attachment D addresses the processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence, and reporting to the NRC. CP&L's response to NRC Item (d) includes a description of the Corrective Action Program and other programs which augment the Corrective Action Program. This includes the Self-Assessment Program, the Independent Assessment Program, the Corrective Maintenance Program, the Procedure Change Program, the Drawing Change Program, the Operating Experience Program, and the Employee Concern Program.

Attachment E provides CP&L's determination of the overall effectiveness of current processes and programs in concluding with reasonable assurance that the configuration of CP&L's plants are consistent with the design bases. In making this determination, CP&L reviewed and considered:

1) Information presented in Attachments A through D, and F;
2) Selected performance indicators; and
3) Management assessments.

Attachment F describes the programs for compiling design bases information at CP&L's three nuclear sites into Design Basis Documents (DBD).

As a result of this effort, CP&L has identified some actions and enhancements to provide additional confidence in the effectiveness of our existing programs. Specific ongoing and future actions discussed throughout the text of the attachments are listed in Attachment G as either a commitment or an enhancement.

Details of CP&L's response are provided in the following attachments:

Attachment A --	Description of engineering design and configuration control processes, including those that implement 10CFR50.59, 10CFR50.71(e), and Appendix B to 10CFR50;
Attachment B --	CP&L's rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures;
Attachment C --	CP&L's rationale for concluding that system, structure, and component configuration and performance are consistent with the design bases;
Attachment D --	Description of processes for identification of problems and implementation of corrective actions, including action(s) to determine the extent of problems, action(s) to prevent recurrence, and reporting to NRC;
Attachment E --	Assessment of the overall effectiveness of our current processes and programs in concluding that the configuration of our plants is consistent with the design bases;
Attachment F --	Description of design bases reconstitution type activities, 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);
Attachment G --	Compilation of commitments and planned enhancements.

Attachment A

Engineering Design and Configuration Control Processes

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Attachment A

Engineering Design and Configuration Control Processes

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I. Introduction

This attachment provides information in response to NRC Item (a):

(a) Description of engineering design and configuration control processes, including those that implement 10CFR50.59, 10CFR50.71(e), and Appendix B to 10CFRPart 50;

Carolina Power & Light Company (CP&L) uses a combination of plant-specific and corporate processes and programs to administer design and configuration control activities. These processes and programs are subject to periodic reviews and assessments and are changed from time to time to strengthen or provide adjustments, as required, to reflect improvements or organizational changes. The discussion which follows is organized into five subsections:

1. **Design Documentation** - types of design information that support design bases information and plant operation;
2. **Configuration Management, Design Control, and Design Change Processes** - programs and processes controlling design and configuration activities including the Engineering Service Request (ESR) process;
3. **Implementation of 10CFR50.59** - the methodology employed to permit changes to the plant without prior NRC approval;
4. **Implementation of 10CFR50.71(e)** - the process of maintaining the Updated Final Safety Analysis Reports (U/FSAR); and
5. **Implementation of 10CFR50, Appendix B** - a description of the Quality Assurance Program.

Where appropriate, efforts to strengthen the programs and processes are discussed. These efforts are also listed in Attachment G as either commitments or enhancements.

II. Design Documentation

As defined in the NRC letter, design bases information is contained in the Final Safety Analysis Report for the Harris Nuclear Plant and the Updated Final Safety Analysis Reports for the Brunswick and Robinson Nuclear Plants (referred to as U/FSAR for each). The design bases information is supported by various types of design documents such as:

1. Design Bases Documents (DBD) containing information for a system, structure, or component (SSC);
2. Calculations such as setpoint and scaling, equipment sizing, and transient analysis;
3. Specifications for procurement and installation activities;
4. Plant drawings;
5. Designated fields in the mainframe Equipment DataBase System (EDBS), such as quality classification and Environmental Qualification (EQ); and
6. Selected vendor manuals.

III. Configuration Management, Design Control, and Design Change Processes

A. Development

Implementing methodologies for configuration management, design control, and design change processes have evolved based upon collective plant and corporate experience, review of industry practices, and commitments to the NRC.

B. Configuration Management

Configuration management refers to the processes and controls established to maintain the physical condition and design bases of the plant. For the engineering organization, configuration management is the process of translating and integrating engineering information into appropriate formats for use by plant organizations in performing their functions in a manner consistent with design bases. This process includes the update of appropriate procedures and training materials to reflect engineering approved changes.

Procedures for implementation of configuration management activities include:

1) Drawing/document prioritization and categorization;
2) Maintenance and control of drawings/documents classified as Category A[1] documents;
3) Updating and issuing of procedures and Category A documents prior to declaring equipment operable;
4) Control of appropriate computer database fields to provide acceptable replacements for "hard copy" documentation; and
5) The design change process that modifies plant configuration and operating or maintenance practices.

C. *Design Control*

CP&L establishes and maintains procedures for design activities such as calculations, drawing changes, and specification changes. Engineering personnel are required to use these approved procedures to translate design inputs[2] into specifications, drawings, procedures, and instructions.

D. *Design Change Process*

Design changes are administered and controlled at CP&L's nuclear plants via the Engineering Service Request (ESR) process. This process is delineated in procedure EGR-NGGC-0005, which provides direction for the initiation, disposition, processing, handling, review, and approval of ESRs including, as required, implementation, testing, and return to service after design changes to SSCs are made at CP&L's nuclear plants. The purpose of this procedure is to maintain design bases integrity and configuration control for CP&L's nuclear plants.

EGR-NGGC-0005 contains requirements, checklists, screening criteria, and/or procedural steps to maintain: the SSCs' design bases (including interfacing SSCs), integrity of plant programs such as Station Blackout, Appendix R (Fire Protection), and EQ, and preservation of organizational and plant operational interfaces and procedures. Temporary modifications[3] (TEM) are also controlled by this procedure.

---

[1] Category A documents are defined in CP&L procedure EGR-NGGC-0007, "Maintenance of Design Documents," as "Those (documents) that are needed to operate and maintain the plant(s) on a real time basis."

[2] EGR-NGGC-0005 defines design inputs as "Those criteria, parameters, (design) bases, or other design requirements, updated to reflect all approved changes, upon which detailed final design is based."

[3] CP&L procedure EGR-NGGC-0005 defines a temporary modification (TEM) as "Temporary alterations made to structures, systems, or components that do not conform with approved drawings or other design documents, and (are) expected to be installed for no more than one fuel cycle."

Figure 1 is a summary flowchart illustrating the design change process. The basic steps in the design change process are:

1) Engineering Service Request Initiation;
2) Initial Screening Process/Assignment;
3) Problem Identification;
4) ESR Design Specification;
5) ESR Development;
6) Review and Approval;
7) Implementation and Testing;
8) Document Updating;
9) Modification Turnover; and
10) ESR Closeout.

1. Engineering Service Request Initiation

An ESR begins as a request for a modification to the physical plant or other work by engineering personnel. ESRs typically involve document changes and/or plant hardware changes. Throughout the process of ESR development, the term ESR is used to describe not only the request, but also the final collection of documents or package used to implement a change or disposition a non-conforming condition. The ESR may be initiated electronically or manually.

2. Initial Screening Process/Assignment

Engineering management is required to screen the ESR to determine if any operability, reportability or corrective action program concerns exist, and to review the ESR for appropriateness, completeness, accuracy, and priority. ESRs may be deleted at this time or at any time in the process, but deletion should consider whether the request maintains or improves the safety, reliability and/or efficiency of the plant. If the ESR is not deleted, the ESR is assigned to an engineering supervisor who then assigns it to a Responsible Engineer (RE).[4]

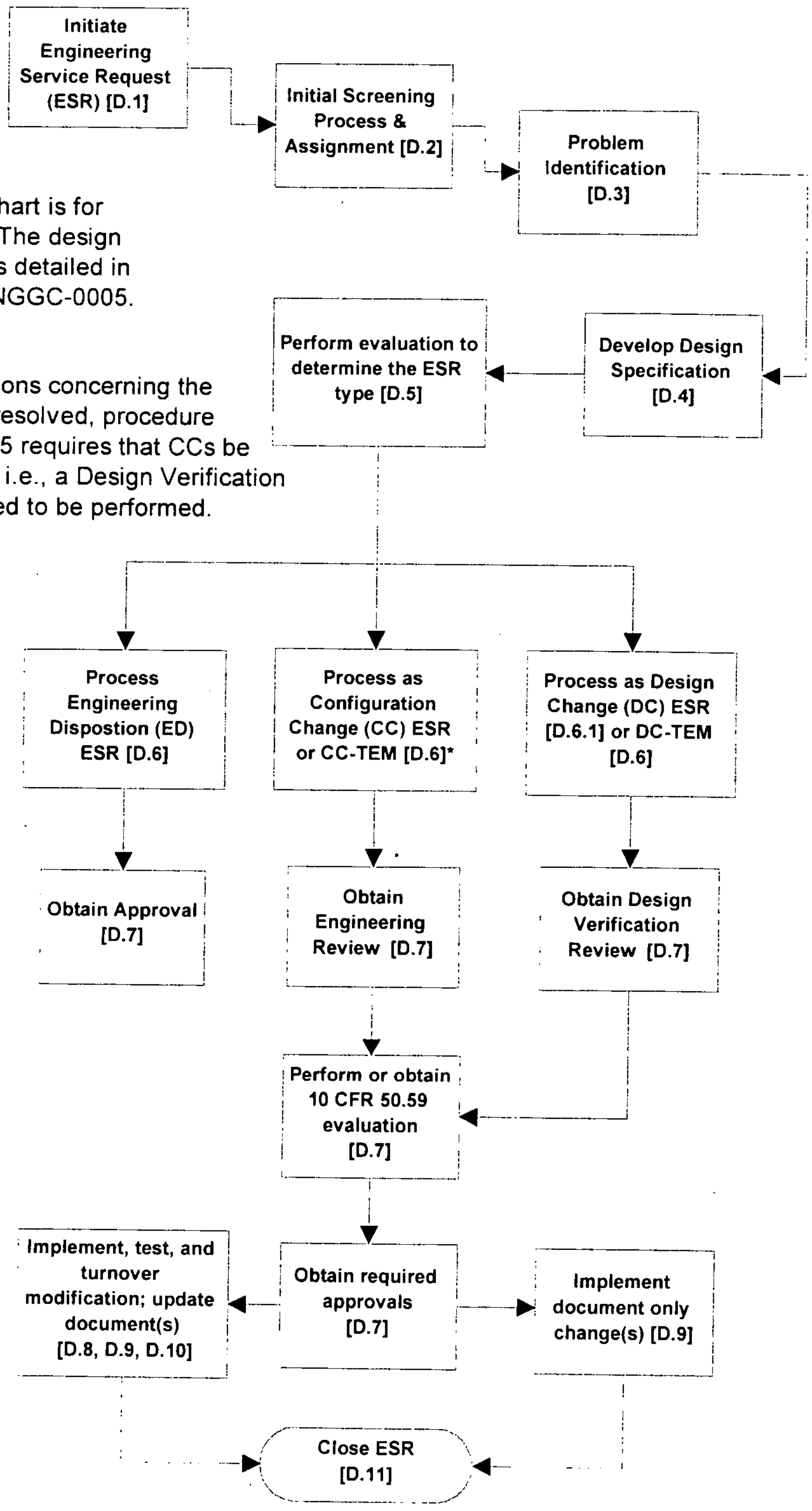
[4] A Responsible Engineer (RE) is defined in EGR-NGGC-0005 as "The individual assigned as the Engineering Organization's single point of contact (for an ESR) and who is accountable for all levels of assigned (design) change activities from problem identification through close-out."

Figure 1

CP&L Design Change Process

Note: This flowchart is for illustration only. The design change process is detailed in procedure EGR-NGGC-0005.

*Until NRC questions concerning the CC process are resolved, procedure EGR-NGGC-0005 requires that CCs be treated like DCs, i.e., a Design Verification Review is required to be performed.



3. Problem Identification

The RE is required to investigate the cause of the problem in sufficient detail to facilitate timely resolution. As a result of this investigation, any operability, reportability, or corrective action program (CAP) concerns are required to be identified and processed. A problem description is required to include, as appropriate: maintenance and operating history, the goal or desired solution, and results of investigation and testing. Based upon the investigation, a proposed resolution and/or response is required. If the proposed resolution or response does not direct or control activities in the plant; change or approve deviations to existing plant configuration, engineering/plant documents, or plant hardware; or produce design outputs[5], the ESR should be processed as an Engineering Disposition (ED). Otherwise, the procedure requires preparation of an Engineering Design Specification.

The RE, in conjunction with engineering management, makes the determination as to the need for a multidiscipline team at this time, or any time in the ESR process, based upon program/organizational impacts, expertise required and/or complexity of the project. The procedure contains screening criteria for use by the RE in determining the need for, and composition of, an ESR team. The ESR team may consist of personnel from engineering, operations, maintenance, and other organizations. ESR team meetings should discuss such items as causal factors, industry experience, alternative solutions, plant risk, key safety functions, radiation protection, performance requirements, constraints, procedure impacts or need for review/revision, and contingency plans. In general, the ESR team is expected to be useful in identifying impacts of design changes on plant programs, operations, and maintenance.

4. ESR Design Specification

Based upon the problem description, the RE should develop and document the functional requirements of the ESR, such as desired performance provisions, in the ESR Design Specification using criteria that include American National Standards Institute (ANSI) Standard N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants" recommendations. The Design Specification is required to document affected design inputs for the SSC, as well as interfaces with other SSCs and associated impacts.

5. Evaluation

An evaluation comparing the functional requirements of the ESR to identified design inputs is required. If the functional requirements require a change to the design inputs, then the ESR is required to be processed as a Design Change (DC). If the requirements

[5] EGR-NGGC-0005 defines Design Outputs as "Documents such as drawings, specifications, and other documents that define the technical requirements of SSCs."

do not require a change to design inputs, the ESR can be processed as a Configuration Change (CC).

6. ESR Development

Three types of engineering information may be created during the design change process. The three types are the Design Change (DC), Configuration Change (CC) and Engineering Disposition (ED) ESRs. A discussion of each ESR type follows including a discussion of ESRs that implement temporary modifications, which are processed as a sub-type of the DC or CC ESR (DC-TEM or CC-TEM).

Design Change (DC) ESR

Based upon the information collected, a response or resolution is required. Affected documents and computer databases are required to be updated and noted on a Drawing/Document Update Form (DUF). U/FSAR changes are required to be documented and processed in accordance with plant procedures.

The ESR Design Section comprises the Problem Identification, ESR Design Specification, and the DUF (in addition to an ESR cover page and a list of effective pages). For ESRs involving modifications to plant hardware, an ESR Installation Section is required in addition to the Design Section. The Installation Section is described in Section III.D.8.

Configuration Change (CC) ESR

The ESR package development for the CC is similar to that for the DC. The difference lies in the type of review obtained of the completed product. The review process is discussed in Section III.D.7.

Engineering Disposition (ED) ESR

Product development for the ED differs from the DC and CC because an ED does not involve a change to design inputs of an SSC or produce design outputs. EDs are used to supply information that interprets or communicates existing design information. An ED may neither independently direct or control activities in the plant nor change or approve deviations from existing plant configurations, engineering/plant documents, or plant hardware.

Temporary Modifications (TEM)

As previously footnoted, TEMs are defined in EGR-NGGC-0005 as "Temporary alterations made to structures, systems, or components that do not conform with approved drawings or other design documents, and (are) expected to be installed for no more than

one fuel cycle." Temporary modification to SSCs may be processed as either a DC ESR (DC-TEM) or as a CC ESR (CC-TEM). TEMs that are implemented to SSCs that are required to remain in service (remain operable) during implementation require processing as a DC ESR.

Category A drawings located in the control room and work control center, that are affected by TEMs, require annotation upon TEM installation and the annotation removed upon TEM removal. Category A drawings may be revised in lieu of annotation as deemed necessary.

The Installation Section of the ESR identifies separate installation and restoration requirements for TEMs. TEM installation and restoration require review by the implementing organization so that necessary measures, such as tagging of equipment, have been implemented correctly.

The expiration date for a TEM is generally limited to the end of the next plant refueling outage, unless approved by the site vice president, or designee. The ESR procedure requires a TEM log be established. This log is normally maintained by plant operations personnel. The RE is requested to periodically audit the log to ascertain whether the log reflects the current status of the TEM.

The procedure identifies the following methods for closure of a TEM:

1) Restoration to the original configuration;
2) Voiding the current ESR and issuing a new ESR; or
3) Conversion to a permanent plant modification.

7. Review and Approval of ESRs

DCs, including DC-TEMs, are used to change the design inputs of an SSC, and therefore require a Design Verification Review in accordance with procedure EGR-NGGC-0003, "Design Review Requirements." This review meets the design verification requirements of ANSI N45.2.11-1974, and is required to include the ESR Design Section and the ESR Installation Section, as applicable.

CCs, including CC-TEMs, do not change the administrative and technical requirements that affect the design inputs of an SSC and therefore do not require a Design Verification Review. An Engineering Review of the complete design package is required to be performed per procedure EGR-NGGC-0003. EGR-NGGC-0003 requires personnel performing an Engineering Review to be qualified in the subject matter and confirm the technical adequacy and accuracy of the proposed design. Please note: until recent NRC questions concerning the Configuration Change (CC) process are resolved, procedure EGR-NGGC-0005 requires that safety-related CCs be treated as DCs (i.e., a Design Verification Review is required to be performed).

The ESR procedure requires a 10 CFR 50.59 evaluation for DC and CC ESRs. This evaluation is to include the ESR Design Section and the ESR Installation Section, as applicable.

As a minimum, DCs and CCs require approval by an engineering supervisor. Temporary Modifications, whether processed as a CC-TEM or DC-TEM, require approval by the Plant General Manager, in addition to an engineering supervisor. Extensions to the expiration date of TEMs require approved by the Plant General Manager and/or the site vice president, or designee.

ED ESRs may not authorize a change; therefore, they do not require a Design Verification Review or an Engineering Review. However, EDs written in support of operability require supervisor approval, and may require a 10CFR50.59 evaluation; otherwise, the RE may approve an ED.

8. Implementation and Testing

For ESRs involving hardware changes, preparation of an ESR Installation Section is required. Plant work planners incorporate this section during development of a modification "work package." The Installation Section includes, as applicable, the following:

1) Installation Summary;
2) Quality Class Determination;
3) Installation Instructions;
4) Installation Sketches (or drawings);
5) Bill of Materials;
6) Spare Parts List; and
7) Modification Testing Requirements.

The Installation Summary requires the RE to address how the ESR request is satisfied, describe system impacts including interfacing or affected SSCs, and such items as precautions, limitations, specific plant conditions (such as outage or on-line), and hold points (where work must stop pending further approvals or inspections). The summary may include any other areas of concern identified during the ESR team pre-installation briefing. Identification of specific implementation phases for modifications requiring partial turnovers (returning control of a part of the affected SSC to Operations) is required.

Performance of a Quality Class Determination, in accordance with plant procedure, is required to identify the quality classification levels of the SSCs involved. This identifies quality class boundaries in support of differing inspection (quality assurance) requirements for modifications to SSCs.

Installation Instructions that are unique to the design, installation, and start-up of the modification are required to be provided in the ESR Installation Section. These instructions shall identify whether sequencing of activities is required for modification implementation, and specify required actions and the responsible individual(s) for tracking completion of the action. As required, Installation Sketches are included. Sketches are required to be based upon present plant configuration as determined via the Nuclear Revision Control System (NRCS) computer database, walkdowns, and/or as-built drawings. A Bill of Materials and a Spare Parts List are provided as required.

Testing Requirements are required to be provided if applicable, including special inspection requirements (methodology, operating parameters and conditions), and quantitative acceptance criteria. This testing is used to determine that the modified SSC performs as designed; the design change has been correctly implemented; the system design bases remain valid; and that Technical Specification limits have not been exceeded.

As applicable, the Installation Section is required to specify the test procedure that will be performed or provide the testing method and acceptance criteria for incorporation into the plant work request. Acceptable testing methods may include inspection (field verification or inspection), demonstration (verified by direct observation), and/or analysis (results determined acceptable following review of data obtained during a test or series of tests). This section of the ESR also requires identification that the specified testing is performed prior to turnover, or whether the testing is performed with the SSC in service (post-turnover). Acceptance criteria (e.g., trip points, reset points, specific functions and ranges including tolerances, and a statement of the system functions to be verified) are required to be provided.

9. Document Updating

Document update requirements are prescribed in procedure EGR-NGGC-0007. In general, Category A documents require updating prior to modification turnover or within ten days of ESR Document Update Notification for non-modification ESRs.

10. Modification Turnover

The ESR procedure requires the RE to track applicable items required for ESR turnover to operations[6]. Items required for turnover may include procedure updates, satisfactory

[6] EGR-NGGC-0005 defines Turnover to Operations as "That milestone in the change process when the modification has been installed, applicable documentation has been accurately updated to reflect the as-installed condition, in-process inspections and verifications have been performed and documented, appropriate post-modification testing has been completed and documented, any open items have been dispositioned, identified as an ESR action item, or otherwise clearly documented and operations has accepted the modified SSC. Turnover may be "partial" or "final" based upon whether work on additional SSCs is ongoing (partial, such as for generic component replacement) or all work is completed (final)."

completion of testing, training of affected personnel or organizations, and updating of affected documents and drawings. If deemed necessary by the RE and/or the ESR team, a walkdown of the installed modification is performed.

11. ESR Closeout

When applicable requirements are complete, the RE may close the ESR. ESR closeout normally[7] occurs within 30 days after final turnover for modification ESRs, or ESR approval for other ESRs.

12. Revisions to ESRs

In general, when changes to an ESR are identified, the revision is processed as the same product type as the original ESR. With the exception of EDs, changes to ESR product types are allowed; however, the RE making the change shall obtain the appropriate reviews and approvals as delineated in EGR-NGGC-0005. Where it is identified that an ED should be revised, and the resolution provided in a DC or CC ESR, the ED is required to be voided.

IV. Implementation of 10CFR50.59

A. Overview

Plant-specific procedures administer the 10CFR50.59 evaluation process as delineated in the corporate guideline titled "Guideline for 10CFR50.59 Safety Evaluations." This guideline is based upon regulatory guidance and information contained in the Nuclear Management and Resources Council (NUMARC) document NSAC-125, "Guidelines for 10CFR50.59 Safety Evaluations."

The 10CFR50.59 evaluation process is used to determine whether a proposed activity may be implemented without prior NRC approval. Figure 2 is an overview flowchart of CP&L's 10CFR50.59 evaluation process.

Activities which are reviewed under the 10CFR50.59 evaluation process may include U/FSAR changes, activities described in an ESR (including the design, implementation, and testing information contained therein), temporary modifications, operating and maintenance procedures, tests, and operation with degraded and/or non-conforming conditions.

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[7] In general, ESR closeout should occur within 30 days as noted; however, there are instances when the ESR may remain open longer than 30 days. For example, a generic ESR that replaces obsolete parts or components on an as-needed basis may remain open longer than 30 days.

The 10CFR50.59 evaluation process consists of two steps. The first step is the Screen, a step to determine whether there is a need to perform the second step, an Unreviewed Safety Question Determination (USQD).

B. *Screening for Applicability*

The Screen is to determine first whether the proposed activity requires a change to the plant Operating License[8] (including the Technical Specifications). If a license change is required, it cannot be implemented under the provisions of 10CFR50.59.

If the proposed activity does not require a change to the plant operating license, and is not bounded by a previously performed 10CFR50.59 evaluation, the individual performing the Screen then answers the following three questions:

1) Does the activity make changes to the plant as described in the Safety Analysis Report (SAR)?
2) Does the activity make changes to procedures as described in the SAR?
3) Does the activity involve a test or experiment not described in the SAR?

If each of the Screen questions are answered "NO," then the activity may be implemented in accordance with plant procedures. It is required that the 10CFR50.59 Screen be reviewed and approved by qualified individuals prior to implementation.

C. *Unreviewed Safety Question Determinations (USQD)*

If any of the Screen questions are answered "YES," plant procedures require a USQD. A USQD documents the responses to the following seven questions:

1) May the probability of occurrence of accidents previously evaluated in the SAR be increased?
2) May the consequences of an accident previously evaluated in the SAR be increased?
3) May the possibility of an accident of a different type than any previously evaluated in the SAR be created?
4) May the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR be increased?
5) May the consequences of a malfunction of equipment important to safety previously evaluated in the SAR be increased?
6) May the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR be created?
7) Is the margin of safety as defined in the bases of any Technical Specification reduced?

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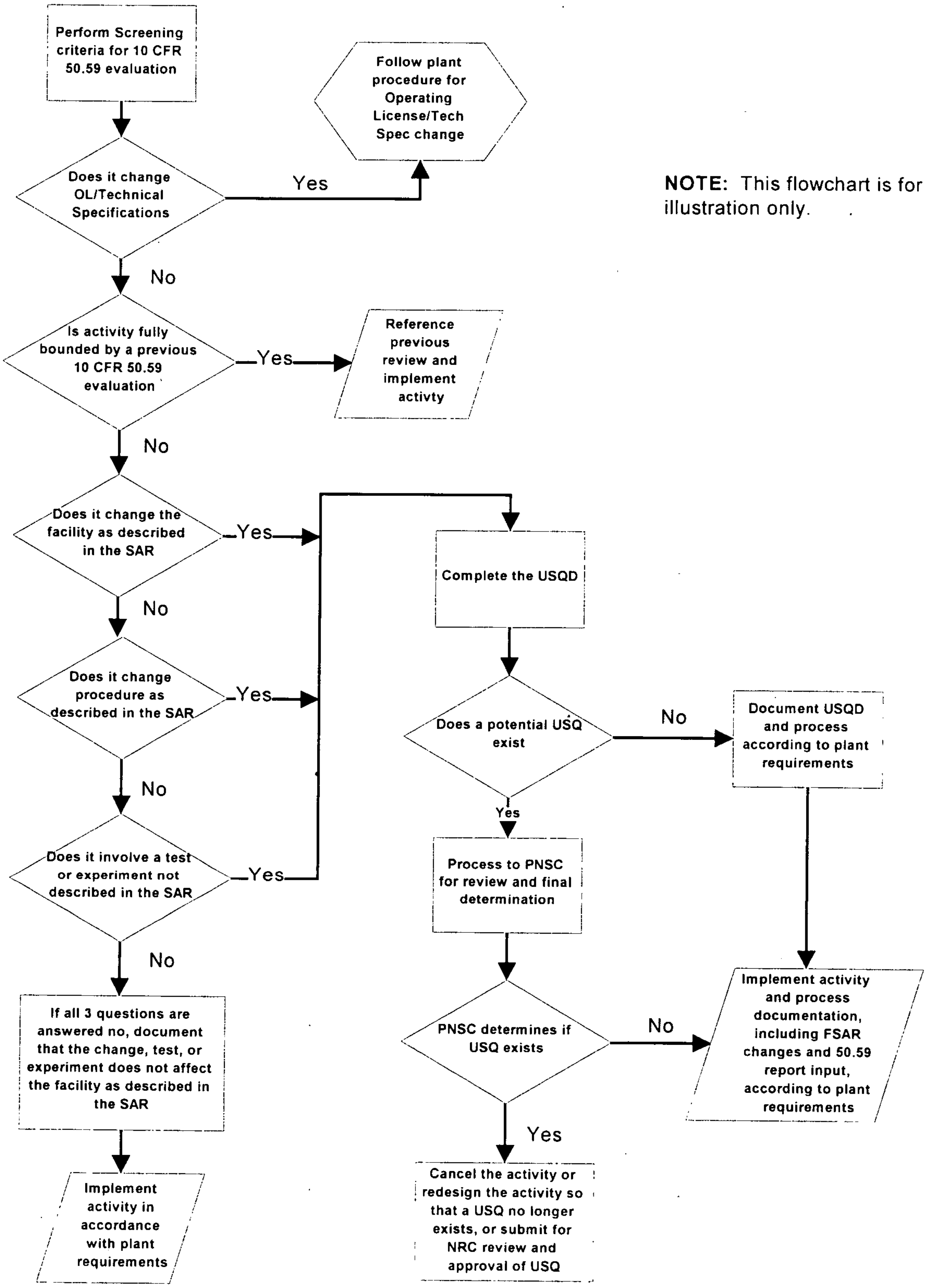
[8] This requirement also applies to the licensing documentation for the Robinson Plant Independent Spent Fuel Storage Installation.

Figure 2
10 CFR 50.59 Process Flowchart



NOTE: This flowchart is for illustration only.

The USQD requires approval by at least two Qualified Safety Reviewers (QSR)[9]. If any question is answered "YES," a USQ may exist. The on-site Plant Nuclear Safety Committee is required to make the determination as to whether the proposed activity involves a USQ. If the proposed activity involves a USQ, the activity cannot be implemented without prior NRC approval.

V. Implementation of 10CFR50.71(e)

The U/FSAR originally submitted as part of the application for the Operating License is required by 10CFR50.71(e) to be updated periodically. 10CFR50.71(e) requires that the U/FSAR be revised to include the effects of:

1) Changes made to the plant or procedures as described in the U/FSAR;
2) Safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve a USQ; and
3) Analyses of new safety issues performed by or on behalf of the licensee at the request of the NRC.

Each site has specific procedures to implement the requirements of 10CFR50.71(e). These procedures require update of the U/FSARs within six months following a refueling outage[10] or annually, and delineate review and approval requirements for U/FSAR changes. In general, differences between the plant specific procedures are administrative.

Updates to the U/FSAR can result from ESRs, 10CFR50.59 evaluations in support of an activity, Condition Reports (CR), editorial corrections due to obvious typographical errors or omissions, or other activities where formal NRC approval may have previously been obtained (such as Operating License and Technical Specification changes).

VI. Implementation of 10CFR50, Appendix B

The Quality Assurance (QA) Program description is in Chapter 17 of the respective U/FSAR. The QA Program provides the administrative basis for enabling conformance with the requirements of 10CFR50, Appendix B. Implementation by plant organizations of the elements of the QA Program relative to engineering design and configuration control activities is provided via the programs and processes described in Sections I through V above.

[9] Qualification as a QSR is based upon education, experience, and training as delineated in plant Technical Specifications and procedures.

[10] Pursuant to an exemption granted by the NRC, U/FSAR updates for Brunswick Units 1 and 2 are required to be submitted within 6 months after completion of Unit 1 refueling outages, but not to exceed 24 months from the last submittal.

A description of appropriate elements of Chapter 17 of the U/FSARs follows below.

A. *Management*

1. Methodology

The QA Program and procedures apply to activities affecting quality such as operation, maintenance, modification, and refueling. The program applies to individuals and organizations responsible for operating and supporting the nuclear plants. The program and procedures define responsibilities and authorities, prescribe measures for the control and accomplishment of activities for the operation of safety-related, fire protection and radwaste systems, structures, and components and requires appropriate verification of conformance to established requirements. A list or system identifying items and activities, to which this program applies, is maintained at each nuclear plant. Controls and responsibilities for maintaining this list or system are prescribed in procedures.

2. Organization

The CP&L organization responsible for safe plant operation is described in Section 13.1 of the U/FSAR and in implementing procedures. The term "line organization" used in the QA Program refers to the production organization reporting to the Executive Vice President/Chief Nuclear Officer.

3. Responsibility

The primary responsibility for quality performance, including the identification and effective correction of problems potentially affecting the safe and reliable operation of the Company's nuclear plants, resides with the line organization.

The managers of functions involving nuclear fuel, engineering, and operations require personnel that are trained for their jobs and that have the experience and education required to carry out their assigned responsibilities. These managers are responsible for providing adequate resources and procedures for implementing the work activities to support this program.

4. Authority

The program and procedures require that the authority and duties of persons and organizations performing activities affecting quality are established and delineated in writing and that these individuals and organizations have been given sufficient authority and organizational freedom to:

1) Identify quality, nuclear safety, and performance problems;
2) Order unsatisfactory work to be stopped and control further processing, delivery, or installation of nonconforming material;
3) Initiate, recommend, or provide solutions for conditions adverse to quality; and
4) Verify implementation of solutions.

5. Personnel Training and Qualification

Both on-site and off-site personnel within the CP&L organization, and contract personnel, who perform activities affecting quality (implement elements of the QA Program), are required to be indoctrinated and trained such that they are knowledgeable and capable of performing their assigned tasks.

6. Corrective Action

The primary goal of the CP&L Corrective Action Program is to improve plant operations and performance by identifying root causes of equipment and human performance problems. Part of this effort is directed toward encouraging individuals to voluntarily report events, near misses, and potential problems. It is the policy of CP&L to seek improvement in each nuclear plant's performance as well as in the performance of supporting departments. The Corrective Action Programs at the sites are discussed in Attachment D.

B. Performance/Verification

1. Design Control

Design changes are subject to controls which are comparable to those applied to the original design. CP&L may designate an organization to make design changes other than the organization which prepared the original design. CP&L and CP&L-designated organizations have access to pertinent background information which can provide the requirements and/or intent of the original design.

Design changes made to the plant are required to be accomplished in a planned and controlled manner in accordance with approved procedures. The plants' Corrective Action Program procedures require errors or deficiencies found in the design process or the design itself be documented and corrected.

Following completion of the design change/modification, controlled design change information is made available to affected personnel. Training, on design changes/modifications that affect the operation of the plant, is required to be provided to affected plant operating personnel.

2. Design Reviews

Procedures identify the types of reviews required prior to implementation of design or configuration changes at the sites. Corporate procedure EGR-NGGC-0003 provides direction on the performance of reviews for design activities at CP&L.

3. Test Control

Procedures provide requirements for test programs and require that items be tested to demonstrate that they will perform satisfactorily in service. Procedures also require modifications, repairs, and replacements be accomplished in accordance with the original design and testing requirements, or acceptable alternatives are provided.

4. Inspection, Test, and Operating Status

Procedures require the identification and control of the inspection, test, and operating status of safety-related SSCs. These procedures require the application, removal, and verification of inspection and welding stamps, or other status indicators as appropriate. Measures are required to be established for indicating the operating status of SSCs. These measures may include the use of checklists, computer programs, logs, stickers, tags, labels, record cards, and test records to indicate the acceptable operating status of installed equipment. Installed equipment which, if operated, could cause damage to other equipment/systems or to personnel are required to be tagged to indicate non-operational status and to prevent inadvertent use.

Selected plant procedures and subsequent revisions receive separate technical review such that required inspections, tests, and other critical operations are included. Altering the sequence of required tests, inspections, and other operations important to safety, should only be accomplished by methods outlined in procedures.

5. Special Process Control

Procedures define requirements for the control of special processes, such as welding, heat treating, and nondestructive examination.

Procedures require that special processes be performed by qualified personnel using proper equipment and in accordance with written qualified procedures. These personnel and procedures are required to be qualified in accordance with applicable codes, standards, and specifications as described in procedures. Qualification records of special process procedures and personnel performing special processes are required to be maintained and available for verification.

6. Inspection

Procedures provide requirements for an inspection program to verify conformance to performance and quality requirements specified for those activities and services. Modification, repairs, and replacements are required to be inspected in accordance with the original design and inspection requirements, or acceptable alternatives are provided.

When acceptance criteria are not met, the condition is required to be documented in accordance with the applicable plant department's corrective action program procedures and reinspected or evaluated, as appropriate.

7. Control of Documents

Procedures provide requirements for the development, review, approval, issue, use, revision, and control of documents. These procedures define the scope of which documents are required to be controlled and identify those individuals or organizations responsible for reviewing, approving, and issuing documents and revisions thereto.

Changes to documents are required to be reviewed and approved by the same organization that performed the original review and approval, or by other designated qualified responsible organizations. Controlled documents are required to be distributed to and used by the person performing the activity in accordance with plant procedures.

8. Records

The QA Program requires that sufficient records are maintained to provide documentary evidence of the quality of items and the accomplishment of activities affecting quality. Procedures outline requirements for the identification, collection, and storage of quality assurance records.

C. *Assessment*

1. Methodology

An objective at CP&L is to encourage ownership, involvement, and dedication by each individual supporting the nuclear plants. This involves looking for ways to improve the performance and safety at each plant. The assessment process, along with the Corrective Action Program, affords a method to identify conditions early and correct them before they become significant quality or safety problems.

2. Self-Assessment

The managers of functions that support the nuclear plants are responsible for implementation of self-assessment activities and processes within their functions on a

continuing basis. Individuals and organizations at the plants are to perform self-assessments. Adverse conditions identified during self-assessment activities are reported and resolved in accordance with the Corrective Action Program.

D. Independent Assessment

1. Independent Assessment Process

The independent assessment process includes gathering data, analyzing data, focusing on selected issues, and identifying deficiencies to desired performance. The results of independent assessments are communicated to management. Management is required to implement actions to correct deficiencies and to develop actions to prevent recurrence. In addition, this process should evaluate corrective measures adopted to eliminate the deficiencies identified.

Based upon plant procedure, data are gathered using performance-based techniques during:

1) Observations of work activities (including line organization self-assessment activities);
2) Interviews;
3) Reviews of documents to gather information (including the use of NRC, INPO, and other agency evaluations);
4) Nuclear Safety Review activities;
5) Team independent assessments; and
6) Analysis of plant data and reports (including adverse condition reports).

Assessment scheduling considerations include the status and safety importance of the activities or processes being performed. The schedule is flexible to allow assessments to be changed depending on plant conditions, events, or issues raised by senior management.

2. Nuclear Assessment Section (NAS)

Each site has a separate section for performing independent assessments. NAS assessments are performance-based and scheduled based upon plant performance and importance to safety, but at a frequency not to exceed 24 months. These assessments should encompass:

1) The conformance of plant operation to provisions contained within the Technical Specifications and applicable license conditions;
2) The performance, training and qualifications of the entire plant staff;
3) The results of actions taken to correct deficiencies occurring in plant equipment, systems, structures or method of operation that affect nuclear safety;

4) The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10CFR50, and
5) Any other area of plant operation considered appropriate by the respective site vice president.

3. Performance Evaluation Support (PES) Unit

The PES, in the Operations & Environmental Support Department, monitors specific functional areas, along with the line organization management, to determine whether the desired levels of performance are being achieved. PES personnel work with each nuclear plant to improve implementation of CP&L's nuclear programs and processes to support safe and reliable operation.

The primary functions of the PES are to:

1) Independently assess the self-assessment and corrective action implementation process of the line organization, and assess the NAS;
2) Share "lessons learned" among the plants and support organizations; and
3) Facilitate the use of industry peer evaluators to identify industry best practices.

The PES evaluates such areas as:

1) The effectiveness of the plant's self-assessment program;
2) Ability to incorporate lessons learned from within CP&L, as well as industry; events; and
3) The corrective action implementation programs.

Attachment B

Translation of Design Bases Into Operating, Maintenance, and Testing Procedures

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I. Introduction

This attachment provides information in response to NRC Item (b):

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

CP&L's conclusions have been based upon plant-specific internal and external assessments of the processes which translate design bases requirements into operating, maintenance, and testing procedures. These processes are subject to periodic reviews and assessments. Thus, they are changed from time to time to strengthen or provide adjustments, as required, to reflect improvements or organizational changes. The plant-specific section in this attachment is organized into five subsections:

A. **Summary** – an overview rationale development;
B. **Procedure Development and Control** – administrative control programs bases;
C. **Assessments and Reviews** – pertinent internal and external evaluations;
D. **Improvement Initiatives** – relevant configuration control improvements;
E. **Conclusion** – rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures.

Where appropriate, ongoing and future efforts which provide, or will provide, additional confidence that design bases information is properly reflected in operating, maintenance, and testing procedures are discussed. These commitments or enhancements are listed in Attachment G.

II. Brunswick Nuclear Plant (BNP)

A. Summary

BNP has demonstrated, particularly in recent years, that programs and processes have been generally effective over the plant life in translating appropriate design bases information into operating, maintenance, and testing procedures. In response to previously identified deficiencies, BNP administrative processes for the preparation, review, and approval of plant procedures have been improved during the life of the plant. These procedures require that proposed new or revised procedures be reviewed to verify that design bases information is incorporated as appropriate. In addition, proposed design changes are evaluated to determine the potential impact on operating, maintenance, and testing procedures. This is reflected, in general terms, by the improved operating performance of the plant over the past few years. The impact of several instances where design bases information was not properly translated into appropriate procedures is discussed.

Several reviews, assessments, evaluations, and inspections which were conducted at BNP during various process improvement programs are described. Internal self-assessments, external inspections, and independent reviews have assisted in identification of problem areas requiring corrective action. Instances of inadequate, incorrect, and omission of design bases requirements in appropriate operations, maintenance, and testing procedures have been identified and corrective action taken. BNP has taken these findings seriously and has responded by assessing the issues, determining the extent of the deficiencies, and taking appropriate corrective actions. These findings were not, however, so significant as to challenge the reasonable assurance that design bases information is generally translated into plant procedures.

The following sections provide additional detail to substantiate CP&L's rationale. In certain areas, CP&L considers that additional reviews or actions will provide further assurance that BNP procedures are consistent with the design bases. These actions are discussed in Attachment G.

B. Procedure Development and Control

The current BNP procedure change program requires that procedures are developed, reviewed, and revised with plant design bases requirements in mind. Some attributes of the program are described in more detail below.

The procedure writer is required to research reference material to prepare or revise a plant procedure. These items include the UFSAR, Technical Specifications, procedure indexes and cross references, NRC regulatory guides and standards, operating experience

feedback items, and outstanding design changes. Several enhancements have increased the confidence that necessary material is identified during the process.

1. Accessibility and Retrievability

The accessibility and retrievability of design bases information for the procedure writer and procedure reviewers has been facilitated by the following tools:

1) Nuclear Revision Control System (NRCS) is an electronic mainframe system that provides current document revision level, status, outstanding changes, documents affected by change documents (e.g., ESRs), cross-reference, distribution profiles and history, and other document information. NRCS is a configuration management tool for identifying, verifying, providing status on cross-referencing, managing, and distributing design documents, change documents, plant operating manual documents, licensing documents, and other documents deemed essential to support the effective maintenance, operation, and configuration control of CP&L's nuclear plants;
2) Computer data base for retrieval of records (STAIRS); and
3) Text electronic search capability as a reference for the UFSAR, Technical Specifications, and industry operating experience.

As described in Attachment A, the current design change process includes requirements for the identification and revision of operating, maintenance, testing and other procedures affected by a modification. There are also programmatic links which identify required procedure changes when license amendments are implemented. Integral to both the modification process and the procedure change process is a requirement to review proposed changes consistent with the BNP 10CFR50.59 program. Implementation of 10CFR50.59 is further discussed in Attachment A.

In 1992, as part of a three-year performance improvement plan, BNP initiated an effort to improve the content of plant procedures and to improve the administrative controls over plant procedures. This effort, completed in 1994, resulted in significant revisions to selected administrative control procedures at BNP. Some of the more significant improvements were: improved control of temporary changes to plant procedures, improved review and approval processes, formal validation of procedures and revisions (including plant walk-downs as appropriate), and a new standard writers guide that factored in the latest available human factors considerations for use when writing procedures.[1] Resultant procedure upgrades were developed and have been, or are being, implemented for system Operating Procedures, Abnormal Operating Procedures, and selected maintenance procedures. While these efforts did not consist of a direct comparison of plant procedures with the design bases, they did provide an overall improvement in document technical quality. The extensive review of plant procedures

[1] This standard writers guide explicitly requires, among other things, that the procedure writer provide that actions authorized by the procedure maintain the plant and components within design requirements.

during this effort also presented many opportunities to identify and correct existing discrepancies in plant procedures.

The processes and controls discussed above provide reasonable assurance that allows us to conclude that design bases requirements are translated into operating, maintenance, and test procedures.

C. *Assessments and Reviews*

As part of continuing efforts to appraise the effectiveness of BNP programs, CP&L has performed assessments for selected systems that included review of operations, maintenance, and testing procedures. These self-assessments addressed, among other things, consistency of the procedures with system design bases. Identified deficiencies require corrective actions to be taken in accordance with the Corrective Action Program. These assessments support CP&L's conclusion that its processes and programs have been generally effective in maintaining plant design bases.

1. HPCI System Safety System Functional Inspection (SSFI)

A self-initiated SSFI was completed on the HPCI system in May 1987. This inspection was conducted by a multidisciplined team using the NRC's guidelines for SSFIs and focused on areas where the NRC had identified weaknesses at other nuclear plants. The stated purpose of the SSFI was to evaluate the HPCI system design bases and identify design and programmatic problems.

The team identified 11 strengths and 11 weaknesses during the review. Weaknesses identified were in the areas of MOV sizing, preventative maintenance activities not performed, inadequate modification evaluation, conflicting and incorrect design information and documentation, procedure errors, vendor recommendations not addressed, lack of specific design bases for some components, and system reliability targets not met. In addition, programs and procedures are currently in place to control many of the areas which had identified weaknesses. BNP has an MOV program which controls MOV motor sizing. Vendor manuals and vendor recommendations (i.e.,Service Information Letters - SILs) are procedurally controlled. Design Bases documents have been reconstituted and maintained.

2. Service Water Operational Performance Inspection (SWOPI)

In June 1994, CP&L performed a SWOPI based upon the NRC Temporary Instruction 2515/2518, Revision 1, and INPO 90-015, "Performance Objectives and Criteria for Operating and Near-Term Operating License Plants." The NRC determined that this effort was sufficient to obviate the need to perform an NRC SWOPI. Two strengths (Service Water system availability/capability, and training), two weaknesses and two issues were identified. The two issues identified were broadly categorized as Engineering

Support and Configuration Control Documentation. Examples supporting the Engineering Support issue included failure to have controls in operating procedures to limit throttling of diesel generator service water outlet valves; failure to implement fully the requirements of Generic Letter 89-13 for Safety-Related Heat Exchangers; failure to control service water inlet temperature to 90 degrees (which is an analysis assumption); failure to consider worst case pump degradation in system hydraulic analysis; and failure to consider loop instrument uncertainty when calculating process limits. These issues were resolved, and corresponding procedures revised to correct the specific deficiencies identified.

The examples of Configuration Control Documentation issues identified included omissions, errors, and oversights in engineering and design documentation including drawings, design bases documents, UFSAR, PRA Analysis, System Descriptions, and calculations. The necessary documents have been revised to address the issues identified.

3. Control Building Heating, Ventilation and Air Conditioning SSFI

This review of the BNP Control Building HVAC System was performed in June 1996. The purpose of the SSFI included the following:

1) Assessment of the Licensee's planned and/or completed actions in response to LER 95-20-01, "CBEAF Unable to Maintain Positive Pressure in the Radiation Protection Mode;"
2) Verification that the CB HVAC system is capable of fulfilling its design requirements and is operated consistent with its design bases;
3) Assessment of the CB HVAC system operational controls, maintenance, surveillance and testing, and personnel training for performance of its safety-related functions; and
4) Maximize the technology transfer to BNP personnel in the conduct of SSFIs.

Weaknesses identified included the lack of completed and approved calculation of record regarding control room habitability due to a Main Steam Line Break (MSLB). Existing calculations included conflicting information regarding control room habitability in a chlorine release, making the determination that the system was consistent with its design bases difficult. The reviewers concluded, however, that the system would have performed its safety function. No significant issues were identified related to translation of design bases requirements into plant procedures. Needed compensatory measures have been put into place, and the resolution of identified issues is being pursued.

4. Electrical Distribution System Functional Inspection (EDSFI)

As documented in NRC Inspection Report 91-09 dated July 3, 1991, the NRC conducted an EDSFI. The report specified several findings and weaknesses in the emergency diesel generator (EDG) and emergency diesel system (EDS) systems. Problems identified included failure to resolve identified problems in a timely manner, and incomplete procedures, calculations and design documents. Weaknesses were identified in BNP

programs for preventative and corrective maintenance on EDGs as well as trending EDG parameters to monitor equipment performance (surveillance issues). A violation (level IV) was identified relating to failure to promptly identify and correct conditions adverse to quality. Errors in UFSAR Sections 8.2.1 and 8.2.2 were noted. Corrective actions were implemented that included management directives, administrative, and programmatic controls.

A 1993 follow-up to the EDSFI is documented in NRC Inspection Report 93-39. The report noted that walk-downs of electrical equipment had been completed as part of CP&L's commitment to update electrical calculations.

5. NRC Violations and Licensee Event Reports

Significant NRC violations and LERs related to translation of design requirements into plant procedures are discussed below.

<u>NRC Violation 91-002-01</u>

Untimely incorporation of vendor manual changes resulted in a CP&L maintenance instruction allowing the use of silicone grease on silicone O-rings in safety-related solenoid valves contrary to the revised vendor manual. Maintenance procedure revisions resulting from vendor technical manual reviews were given a higher priority, and subsequently a new Vendor Manual Program was implemented in the engineering organization.

<u>LER 2-93-003</u>

The Drywell Spray Outboard Isolation Valve body was installed in the reverse direction which resulted in the associated body drain being inside the primary containment boundary. The situation was identified in 1974, and deemed to be acceptable based upon the determination that the isolation valve would perform its design function. However, this configuration resulted in the associated body drain being inside the primary containment boundary, which was not recognized. The appropriate design documents were not revised to show the installed configuration, and the drain valve was not included in Technical Specification leak rate testing. A review of other PCIS globe valves was performed and did not reveal any deficiencies.

D. Improvement Initiatives

1. System Operating Procedures

System Operating Procedures (OPs) have been upgraded to incorporate human factor improvements provided by the plant Generic Writers Guide. The upgrade also provided an additional review of the technical accuracy of the procedures. These revisions

incorporated corrective actions from ESRs, Action Items, and Procedure Action Requests (PARs). This program included, for example, Control Room panel walk-down to verify that control switch labeling was correct. This effort contributed to the removal of electrical and valve component duplications in procedures, which will aid in maintaining accurate configuration control.

2. Electrical Load Lists

Load lists have been developed for plant electrical distribution panels. In this effort, completed in 1996, design information for electrical loads was confirmed to be reflected in procedures. A physical walk-down of electrical distribution panels was performed on the Emergency buses and uninterrupted power supply (UPS) with other buses receiving validation with prints, discrepancy resolution, and modification research. Follow-up activities included print updates to reflect field conditions and plant OP lineup updates for consistency with the load lists. Other human factor enhancements were identified and made as a result of operator validation of each procedure. This project provides increased confidence in the accuracy of plant procedures, as well as making electrical load information readily available to plant operators.

3. Maintenance Procedures

Beginning in 1983 and continuing through the end of 1996, Maintenance has been implementing several projects to improve and upgrade the technical quality of its procedures. Efforts were made to verify the correct translation of the design bases into plant procedures. In 1983, a contract was entered into with General Electric (GE) to provide support for developing a Procedure Writers' Guide and convert approximately 1400 maintenance instructions to maintenance procedures. This effort evolved into several upgrade projects including Maintenance Surveillance Test Upgrades, Single Failure Reviews, PM program optimization, the Regulatory Required Instrument List (RRIL) review and required procedure changes, and format enhancements. During the procedure conversion phase of this effort, GE (and subsequent) contractors, under CP&L supervision, reviewed reference documents to verify the technical accuracy of the information contained in the procedures.

The current project being performed to bring maintenance procedures into compliance with the administrative controls procedure described above will incorporate any outstanding revision requests, as well as perform validation of procedures where required. This project is due for completion by September 1, 1997.

4. Emergency Operating Procedures (EOPs)

BNP EOPs are based upon BWR Owners' Group Emergency Procedures Committee guidance that has been approved as generally acceptable for implementation by the NRC[2]. CP&L verified and validated[3] these procedures.

Design bases values contained in the EOPs are specifically identified in the PSTG/EPG comparison packages. These packages contain discussions of the values used in the EOP and provide references to relevant source documents. Calculations performed for the EOPs are documented in calculation packages identified by "0EOP-WS-xx" or "0EOP-PSD-xx" series numbers. Sources of information for these packages include the UFSAR, Technical Specifications, and other plant reference documents.

In 1991, Operations Engineering, Inc. (OEI) performed an independent audit of the BNP EOP procedures and related processes. OEI had participated in the development of earlier EPG revision strategies. The audit activities included a detailed review and evaluation of:

1) The technical accuracy and completeness of the EOPs and associated supporting documents;
2) The adequacy of the defined scope and prescribed sequence of programmatic activities that are specified in applicable procedures for the development, review, approval, implementation, and maintenance of the EOPs; and
3) The completeness and correctness of written records documenting the performance and completion of EOP development, review, approval, and implementation activities consistent with programmatic requirements specified in applicable procedures.

The audit team reported that the EOP development process was well structured, and that the scope of activities audited appeared to comply with NRC requirements. Weaknesses were identified with documentation of the basis for certain elements of the PSTG and the EOPs, as well as with the EOP validation and verification procedure. Minor calculation errors were also identified.

In 1994, CP&L performed a self-assessment of the EOPs to determine whether the procedures, as written, perform their intended function for emergency operation. Items were identified in the areas of generic labeling, administrative, technical, operational, and the writer's guide. These items have been dispositioned under the Corrective Action Program (CAP.)

5. Inservice Inspection Program/Inservice Testing Program

[2] BWR Owners' Group (BWROG) Emergency Procedure Guidelines (EPGs), Revision 4.

[3] Guide 0OI-37," Preparation and Review of the Plant Specific Technical Guideline," documents the acceptability of the conversion of the BWROG EPGs to Brunswick's Plant Specific Technical Guidelines (PSTGs). This effort is documented in Attachment A to 0OI-37, with Attachment C to 0OI-37 providing the justification for any differences or deviations of the PSTGs from the EPGs.

Administrative controls for the Inservice Inspection (ISI) program of the Brunswick Nuclear Plant (BNP) are described in procedure 0ENP-16, "Procedure for Administrative Control of Inservice Inspection Activities." The administrative controls for the Inservice Testing (IST) of pumps and valves are included in procedure 0ENP-17, "Pump and Valve Inservice Testing." These programs implement requirements of the Code of Federal Regulations, American Society of Mechanical Engineers Section XI, North Carolina Boiler and Pressure Vessel Code, and BNP Technical Specifications.

The Brunswick Three-Year Plan initiative TY-507, "Inservice Inspection / Inservice Testing Improvement Program" was developed in 1992. The objectives of TY-507 were:

1) Improvement of the program, including development of a bases document;
2) Enhance training of personnel through code seminars, technical courses; and
3) Development of administrative controls for processes so that ISI/IST programs are properly implemented and maintained in compliance with regulatory requirements.

This initiative identified a number of deficiencies in the ISI/IST program, including untimely completion of required testing, inadequate test sequencing, non-compliance with generic letter requirements, and non-compliance with relief request restrictions. Corrective actions were implemented and completed for issues identified.

6. Power Uprate/Improved Technical Specifications

CP&L is currently implementing its Improved Technical Specifications program for BNP. Because this activity, by definition, involves the revalidation of key design bases-type information, it will provide, when completed, increased assurance that plant procedures are consistent with design bases.

CP&L is presently in the final stages of implementing the Unit 1 Thermal Power Uprate project, with the Unit 2 project to follow in the fall of 1997. The broad scope of systems and programs affected by this modification afforded the opportunity to perform in-depth reviews of design bases information contained in numerous analyses, calculations, and plant procedures. Affected documents were identified and revised as required by the design change process. Approximately 200 plant procedures were revised as a result of this modification.

During CP&L's review of design information, a potentially significant error in the power uprate data was identified. As was reported to the NRC, an error was made in the peak suppression pool temperature calculation for a postulated Station Blackout event. The corrective actions are to:

1) Revise the UFSAR and DBDs so that the containment analysis inputs and acceptance criteria are clearly defined; and

2) Perform an evaluation of the verification and validation practices for outsourced engineering.[4]

Another lesson learned from the power uprate review related to difficulties with identifying and locating calculations that were potentially affected by power uprate. Issues with organizational ownership, and categorization and control of calculations were identified as barriers to accessing the necessary calculations. These issues are presently being dispositioned in the Corrective Action Program.

7. Generic Letter 96-01 Surveillance Test Procedure Review

On January 10, 1996, the NRC issued Generic Letter (GL) 96-01, "Testing of Safety - Related Logic Circuits." CP&L responded to the NRC that BNP was committed to addressing the problems identified in the GL and to implement actions requested by the GL. The requested actions include comparison of electrical schematic drawings and logic diagrams for the Reactor Protection System, EDG load shedding and sequencing, and actuation logic for the engineered safety features systems against plant surveillance test procedures to ensure that the logic circuitry, including the parallel logic, interlocks, bypasses and inhibit circuits, is adequately covered in the surveillance procedures to fulfill Technical Specifications requirements. Surveillance procedures will be modified as necessary.

To support this initiative, a team has been assembled including a project manager and representatives from operations, maintenance, regulatory affairs, and engineering. The performance and documentation of this effort will be governed by Special Procedure 0SP-97-001, "Guides For The Performance Of Technical Specification Surveillance Reviews For Safety-Related Logic Circuits," which is presently in the approval process. The actions required by GL 96-01 are scheduled for completion prior to the startup from the first refueling outage commencing one year after the issuance of the letter. These outages are: the Unit 2 outage (B213R1) scheduled for the 3rd Quarter of 1997, and the Unit 1 outage (B112R1) scheduled for the second quarter of 1998.

These initiatives and upgrades improved the technical quality of the target programs and procedures, and helped verify linkage to design requirements (as discussed above).

E. Conclusions

The rationale for concluding with reasonable assurance that design bases requirements are adequately translated into operating, maintenance, and testing procedures is based upon the following key elements:

1) Established processes for procedure development and control;

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[4] These corrective actions were provided to the NRC in CP&L letter dated December 23, 1996.

2) Programmatic links for identification of affected procedures in the design change or license change processes;
3) Accessibility of design information;
4) Implementation of 10CFR50.59;
5) Corrective action program; and
6) Self- and third-party assessment programs.

Deficiencies have been periodically identified in the operation, maintenance and test procedures. However, when their number, type, safety significance, and resulting corrective actions are considered in the whole, these deficiencies do not change our conclusion today. CP&L will continue to periodically assess processes and procedures and make changes, as necessary, for continual improvement.

III. Harris Nuclear Plant (HNP)

A. *Summary*

Controlled processes were established for development and review of the original HNP procedures. Procedure development and revision, design change, and configuration management processes were put in place prior to initial plant startup. Currently, there are approximately 3,800 controlled procedures which direct operation, maintenance, and testing activities at HNP. The current procedure generation, revision, review and approval process provides additional confidence that design bases information is correctly translated into operation, maintenance, and test procedures.

In accordance with plant procedures, plant processes and programs are subject to periodic assessments. Adverse trends in process or procedure adequacy are identified and corrected using the corrective action program described in Attachment D. Moreover, experience with the use of these procedures provides additional confidence that use of these procedures keeps the plant systems operating within the design bases.

Although some procedure deficiencies regarding design bases issues have been identified, it is our judgment that the issues are not so significant as to challenge the overall confidence in CP&L's ability to operate, maintain, and test in accordance with the design bases. Some examples of these deficiencies, and the subsequent corrective actions are described in this attachment.

Accordingly, CP&L concludes that the procedure control processes provide us with reasonable assurance that design bases requirements are adequately translated into operating, maintenance, and testing procedures. The following sections provide additional detail to support this rationale. In certain areas, CP&L considers that additional reviews or actions will provide further assurance that HNP procedures are consistent with the design bases. These actions are discussed in Attachment G.

B. *Procedure Development and Control*

1. Original Procedure Development

Documentation available for use during original operating, maintenance, and test procedure development in the 1983-1984 time frame included drafts of the Technical Specifications and system descriptions, as well as the FSAR, applicable industry standards, technical manuals, and controlled drawings. Pre-operational walk-downs were also a method available for development of the original operating, maintenance, and test procedures. These procedures were developed and approved in accordance with Plant Procedure, PP-3, "Procedure Review and Approval," which provided standardized instructions for review and approval of new procedures. This procedure was intended to

ensure the procedures were in compliance with 10CFR50 Appendix B Criteria VI, ANSI N18.7, and the HNP Technical Specifications. In addition, the Procedures Administrative Manual, (PAM), was established in June 1983 to provide a uniform system of Plant Operating Manual procedure organization, administrative controls, and procedure development for CP&L nuclear power plants.

The original Emergency Operating Procedures were developed using the Westinghouse Owner's Group Emergency Response Guidelines (ERGs) and Writers' Guide. Step Deviation Documents were prepared to justify deviations from the Guideline.[5] Simulator testing, table top reviews, and control room walkthroughs were performed as part of the EOP validation and verification program.

2. Current Procedure Control Process

The current HNP procedure development and change process is described in plant procedures. The procedures require the writer to research necessary material required to prepare the procedure. These items include:

1) FSAR;
2) Technical Specifications;
3) Procedure indexes and cross-references;
4) NRC regulatory guides;
5) Standards;
6) Operating experience feedback items; and
7) Outstanding design changes.

The procedures contain requirements for administrative and technical reviews. The procedures also require that 10CFR50.59 evaluations be completed for changes to operating, maintenance, and test procedures. See Attachment A for a description of the 10CFR50.59 process. Procedure approval authority is controlled by established plant procedures.

As described in Attachment A, the current design change process includes requirements for the identification and revision of operating, maintenance, testing, and other procedures affected by a modification. There is a single point of accountability (responsible engineer) for a modification. This is intended to strengthen communications between the design engineer and the procedure writers. The modification process requires that these procedure revisions be completed prior to modification turnover for operation.

[5] A Step Deviation Document documents how the Westinghouse Owners' Group Emergency Response Guidelines (ERGs) are incorporated into the EOPs on a step-by-step basis and provides justification to any deviations from the ERGs.

In addition, procedural controls are established to identify required procedure changes when Operating License amendments are implemented. Implementation plans for incorporating these amendments require approval by the Plant Nuclear Safety Committee (PNSC).

3. Accessibility and Retrievability

The accessibility and retrievability of design bases information for the procedure writer and procedure reviewers has been facilitated by the following tools:

1) Nuclear Revision Control System (NRCS) is an electronic mainframe system that provides current document revision level, status, outstanding changes, documents affected by change documents (e.g., ESRs), cross-reference, distribution profiles and history, and other document information. NRCS is a configuration management tool for identifying, verifying, providing status on cross-referencing, managing, and distributing design documents, change documents, plant operating manual documents, licensing documents, and other documents deemed essential to support the effective maintenance, operation, and configuration control of CP&L's nuclear plants;
2) Computer data base for retrieval of records (STAIRS); and
3) Text electronic search capability as a reference for the FSAR, Technical Specifications, Plant Operating Procedures, and industry operating experience.

The established processes for procedure development and control, the programmatic links for identifying affected procedures in the ESR or Operating License amendment processes, the accessibility of design information, and implementation of 10CFR50.59 are elements of CP&L's rationale for concluding with reasonable assurance that design bases requirements have been adequately translated into operating, maintenance, and test procedures.

C. *Assessments and Reviews*

Self-assessments, independent assessments, Licensee Event Reports, and Notices of Violations were reviewed to identify those events involving significant inconsistencies between design documentation and procedure content. Some recent examples are summarized in the following paragraphs.

1. Self-assessments

Scaling and Maintenance Procedures

In 1994, an assessment of the scaling[6] and maintenance procedures program was conducted. The stated purpose of this assessment was to evaluate the effectiveness of

6 Scaling documents are used to select adjustments to process instrumentation to convert correctly plant variables (e.g.,pressure and level) from engineering units (e.g.,psig, and inwc) to analog voltage or electrical current units.

incorporating design changes and/or engineering setpoints into plant equipment. Twenty-two (22) maintenance procedure revisions were reviewed to determine whether scaling document revisions for Refueling Outage 5 (RFO5) setpoint changes were correctly incorporated. The assessment methodology included interviews to determine the reasons for the perception of poor scaling document quality. No significant safety concerns were identified.

Although no error was found in the 22 maintenance procedures, three weaknesses were identified. The weaknesses involved the lack of a procedure to provide a standardized process for incorporating scaling document revisions into appropriate maintenance procedures; a lack of a cross-reference from the scaling document to applicable maintenance procedures; and a lack of formal training on process instrumentation scaling principles and card operation. Corrective actions were taken. Scaling document revisions are now processed by the ESR program described in Attachment A. The scaling documents are now to be included in the NRCS.

Motor Operated Valve Program

A self-assessment to evaluate the effectiveness of the HNP Motor Operated Valve (MOV) program was performed in November 1994. The scope of the assessment included a review of the interface between the MOV program and the design change process. In addition, the assessment included a review of MOV test procedures to determine if design requirements were adequately reflected. The design change process was evaluated to verify that controls were in place to identify potential impact on MOVs within the program scope. Forty (40) engineering ESRs were reviewed to evaluate the effectiveness of the design change process in identifying work scopes which could potentially impact the MOV program. The review indicated that the ESR screening process is effective in identifying items which may impact the MOV program. As part of the assessment, the acceptance criteria for periodic testing were reviewed to verify that design requirements are adequately reflected. To assess this design transfer process, eleven (11) MOV setup calculations were reviewed for consistency with design bases information. No significant problems in this area were identified. Overall, the MOV self-assessment did not identify any specific program weaknesses; however, recommendations were made to strengthen the program administrative controls. The results of this MOV self-assessment support the rationale that design bases requirements have been properly translated into plant testing procedures.

Procedure Revisions Resulting from Modifications

In late 1995, an assessment was performed to determine whether the Plant Operating Manual procedures were being properly revised to incorporate plant modifications. This assessment included an evaluation of selected modifications to determine whether affected procedures were identified and subsequently revised to reflect the design changes. Six plant modifications, including Plant Change Request (PCR) 6502, "AFW

Flow Control Valve Auto-Open," were reviewed. These modifications involved actuation circuitry and were implemented between September 1992 and October 1995.

One issue identified in this self-assessment was that some procedures requiring revision were not identified by the modification and/or review process. The assessment identified approximately 100 procedures which should have been revised as a result of the PCR 6502. Many of the identified procedure revisions were to alert operators of automatic open signals in procedures where an actuation signal would not be expected. An engineering ESR (ESR 95-01041) has been initiated to track necessary procedure revisions.

Another issue identified in this assessment was that the acceptance test for the modification did not test each required signal. The appropriate test procedure was developed and the required testing performed. An item was added to the design change screening checklist to require a review of response time and overlap testing if a modification involves actuation circuitry.

During the review of the other five modifications, additional, less significant configuration-related issues were identified. These issues were resolved through the Corrective Action Program.

An engineering management review of the identified procedure inadequacies concluded that the findings themselves did not involve significant safety consequences. It was further concluded that the procedure implementation problems for PCR 6502 were not typical of other modifications implemented at HNP due to the modification's complexity, and the large number of potentially impacted procedures.

Self-Assessment of the 10CFR50.59 Program

A self-assessment of the 10CFR50.59 implementation process was conducted in April 1996. The purpose of the self-assessment was to determine whether the 10CFR50.59 program conforms to the Code of Federal Regulations with primary focus on the quality of the safety evaluations. The assessment concluded that the program generally complied with the requirements of 10CFR50.59. One programmatic weakness was identified in that certain types of procedures had been exempted from 10CFR50.59 evaluation requirements. These exemptions were revoked and the procedures were evaluated in accordance with the 10CFR50.59 process. Additional programmatic enhancements were also identified.

The results of NAS and Licensing Unit day-to-day review of 10CFR50.59 evaluations indicate that additional training should be conducted for qualified safety reviewers. This training will be conducted as described in Attachment G.

Safety System Functional Evaluation of the Reactor Trip System

A safety system functional evaluation of the Reactor Trip System was performed in December 1996. The evaluation was intended to assess the adequacy of engineering activities, such as modifications, evaluations, and calculations, to verify that these activities did not inadvertently alter the plant design bases and operational requirements, and to assess and evaluate the conformance to the applicable FSAR sections. As part of this evaluation, a review of the Manual Reactor Trip and Reactor Coolant Pump Bus Undervoltage Trip functions was performed to determine if there was sufficient overlap and comprehensiveness in the logic testing procedures. This evaluation identified examples of procedure deficiencies. The examples included a maintenance procedure that had not been updated when a reference was updated and procedures that had not been updated in conjunction with a plant modification. A review of the issues identified in the self-assessment report is in progress, and corrective actions are under development.

2. Independent Assessments

Nuclear Assessment Section (NAS) Biennial Procedure Review Assessment

In 1993, HNP committed to have NAS conduct a biennial review of three to five percent of routine plant procedures that are used more frequently than once every two years. The purpose of this review was to evaluate the acceptability of the procedures and to verify that the procedure review and revision programs are effectively implemented. The first biennial procedure review assessment conducted by the NAS was completed in July 1995. One hundred and twelve procedures were reviewed. Although deficiencies in other areas were identified, the assessment verified that procedures matched the installed systems and that modifications and other design changes had been accurately incorporated. No finding in the area of design bases was identified.

NAS Engineered Safety Features Response Time Testing Assessment

An assessment of the "A" Train Engineered Safety Feature Response Time surveillance was conducted by the NAS in late 1995. This assessment identified issues involving inconsistencies between the FSAR and PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," and a deficiency involving inadequate screening during the modification process for potential impact on Engineered Safety Feature response time testing. Specific to the finding, the Engineering I&C/Electrical Screening Checklist was modified to require a review of response time and overlap testing if a modification involves changes to actuation circuitry. In order to address the overall FSAR/design consistency issue, an action plan was developed to evaluate the present condition of the FSAR and make any necessary revisions, and to improve programmatic controls to maintain future FSAR fidelity. The FSAR improvement plan is discussed in Attachments C and G.

3. Licensee Event Reports/NRC Notices of Violation

Failure to Translate Design Information into Procedures (NOV 91-15-01, LER 91-010)

One Level III violation was cited, on July 23, 1991, for failure to maintain two operable automatic trip instrumentation channels when the Reactor Trip System breakers were closed and the Control Rod Drive System was capable of rod withdrawal. This condition was identified subsequent to a June 3, 1991, reactor trip in which the A Reactor Trip Breaker failed to open automatically. This violation occurred because the appropriate precautions were not added to maintenance procedures used in troubleshooting or testing of the reactor trip breakers upon issuance of a Westinghouse Technical Bulletin. This is considered an example of a failure to properly translate design bases information (i.e., the Westinghouse Technical Bulletin) into plant procedures. Corrective actions included replacing the damaged reactor trip breaker undervoltage driver card, completing additional reactor trip switch testing, and changing reactor trip breaker maintenance and test procedures.

Inadequate Self-Checking and Technical Reviews Led to Error in Procedure (LER 96-004 and NOV 96-02-02)

On February 6, 1996, an inoperable level transmitter was erroneously removed from bypass and placed into service due to an error in procedure. This condition was the subject of a Licensee Event Report and a Level IV Notice of Violation. The procedure error had been introduced during a 1995 revision. The cause of the error was determined to be inadequate self-checking during the revision process so that the intended action was correct and not referencing the applicable drawings. No interdisciplinary review was performed on the revision. Corrective actions included revising the procedure and identifying procedure changes which require an engineering review prior to approval.

Ventilation System Surveillance Testing (LERs 96-007 & 96-009)

On April 25, 1996, it was determined that the procedure for performing surveillance testing on the Control Room Emergency Filtration System was incomplete in that it did not measure the differential pressure for the adjacent areas as required. The cause of this procedure problem was inadequate procedure development due to an incorrect interpretation of necessary testing requirements. One of the corrective actions for this condition was a review of the Reactor Auxiliary Building Emergency Exhaust System (RABEES) surveillance testing.

On May 30, 1996, deficiencies were identified in the RABEES surveillance test. The differential pressure for enclosed pump rooms was not being measured during the test. Another test deficiency identified was that the operation of some non-safety fans was discovered to be aiding the RABEES in maintaining the required differential pressure. These non-safety fans may not be available following a loss of offsite power incident.

These inadequate testing procedures were a result of incorrect interpretation of Technical Specification testing requirements during initial procedure development. The testing procedures were revised. As stated in the LERs, these conditions were determined to have minimal safety consequences.

Surveillance Test Deficiencies Causing TS 3.0.3 Entries (LER 96-010)

On June 14, 1996, a deficiency was identified in the quarterly Residual Heat Removal System (RHR) surveillance test procedures. In the test, the operable RHR train was cross-connected with the inoperable RHR train being tested. The cause of this inadequate test procedure was determined to be personnel error during procedure revision. Subsequently, an additional test deficiency was discovered in the surveillance test for the Containment Ventilation Isolation Area Radiation Monitors. The test, as written, caused both trains of the Containment Vacuum Relief System to become inoperable. This inadequate testing procedure was a result of an incorrect interpretation of Technical Specification test requirements during initial procedure development. Both test procedures were revised. One corrective action for this event was a review of a sample of procedures to identify any similar deficiencies. This follow-up review did not identify other procedures with similar deficiencies.

Interface Boundary Issues Between Safety and Non-Safety Components (LER 96-013)

On August 1, 1996, using Operating Experience Feedback information from another plant, Operations personnel identified that HNP procedures permitted the Refueling Water Storage Tank to be connected to the non-seismically qualified portions of the fuel pool purification system and to a non-seismic hydrostatic test pump. This condition was caused by a failure to reconcile operating procedure lineups with plant design bases during original procedure development. A review of other seismic/non-seismic interface boundary valves for similar problems was completed. Three additional examples of interface boundary issues were identified. Administrative controls were established to maintain the seismic boundary valves closed. Procedures were revised so that the valves are not operated when the safety system is considered operable.

Although some of the examples in this section have been identified by a third party, many of the procedure problems relative to design bases have been self-identified by self-assessments, root cause investigations for adverse conditions, and reviews of operating experience information. This evidence of an effective corrective action program provides additional support for a conclusion that design bases information is adequately reflected in plant operating, maintenance, and test procedures.

D. Improvement Initiatives

Regulatory Guide 1.33 Appendix A Review

HNP identified in 1995 that adequate corrective maintenance procedures did not exist for some activities described in Appendix A of Regulatory Guide 1.33. This was identified during a root cause evaluation of repetitive maintenance problems with a Feedwater Isolation Valve which also resulted in a Notice of Violation. Corrective actions included reviewing other maintenance activities which could affect the performance of safety-related equipment and identification of those components requiring detailed procedures. New procedures were developed for corrective maintenance of 20 components.

The Operations, Environmental and Radiological Control, and Engineering organizations also reviewed Regulatory Guide 1.33 and compared the list of activities requiring written procedures to activities performed within their organizations. No additional Regulatory Guide 1.33 activities were identified for which procedures did not exist.

Review of Surveillance Testing of Safety-Related Logic Circuits

A systematic review to verify the adequacy of the existing Technical Specification surveillance procedures for plant actuation logic testing associated with the Reactor Protection System, Emergency Diesel Generator load shedding and sequencing, and Engineered Safety Features components was completed in late 1996. Although this review is being used to satisfy Generic Letter 96-01, "Testing of Safety-Related Logic Circuits" requirements, it was initiated in the fourth quarter of 1995 based upon an adverse trend regarding surveillance procedures. Procedures for sixteen actuation circuitry subsystems were reviewed. As reported in LER 96-02, the review identified testing omissions introduced during initial procedure development. The review also identified surveillance procedure deficiencies introduced during the procedure revision process.

The causes of the procedure inadequacies were a lack of understanding of overlap testing requirements, a lack of engineering/system expert involvement in the procedure development, improper interpretation of technical specification requirements, and validation of individual technical specification line-item requirements rather than complete safety function validation.

Corrective actions included performing missed surveillances and implementing engineering review of Reactor Protection System and Engineered Safety Features surveillance procedure revisions. Planned actions include two modifications to correct design deficiencies, revisions of additional procedures, training for appropriate procedure writers, and a surveillance program review as part of the conversion to ITS.

FSAR Improvement Plan

HNP began an FSAR Improvement Initiative in May of 1996. This review has resulted in the identification of testing discrepancies. As described in LER 96-016, it was determined that the FSAR specified that the reactor trip bypass breakers would be tested prior to being placed in service. However, previous testing had allowed the breakers to be racked fully into service prior to testing. Appropriate procedures were revised to require testing of the reactor trip bypass breakers prior to being placed in service.

Surveillance Program Review Project

A project is underway to review surveillance procedures against the requirements of Technical Specifications. Elements which will be considered in the review include overlap testing, test condition limitations, and ability of testing to provide assurance that design functions will be met even in the most limiting conditions.

These types of completed and planned initiatives are evidence that HNP is committed to continual improvement in the area of procedures. These types of initiatives will continue to be undertaken, as necessary, in response to adverse trends identified by the corrective action program, assessments, or industry events.

E. Conclusions

The rationale for concluding with reasonable assurance that design bases requirements are adequately translated into operating, maintenance, and testing procedures is based upon the following key elements:

1) Established processes for procedure development and control;
2) Programmatic links for identifying affected procedures in the design change or Operating License amendment processes;
3) Accessibility of design information via NRCS;
4) Implementation of 10CFR50.59;
5) Effective self and independent assessment programs; and
6) Corrective action program.

Deficiencies have been periodically identified in the operation, maintenance and test procedures. However, when their number, type, safety significance, and resulting corrective action are considered in the whole, these deficiencies do not change our conclusion today. CP&L will continue to periodically assess processes and procedures and make changes as necessary for continual improvement.

IV. H. B. Robinson Nuclear Plant (RNP)

A. *Summary*

Robinson Nuclear Plant (RNP) was originally designed and constructed as a turn-key project in the late 1960s and has been conducting operations for 25 years. The plant was placed in-service using a set of custom developed operations technical specifications and procedures appropriate for the time period.

RNP has operated during much of the evolutionary history of commercial nuclear power in the U. S. Consequently, as standards for procedure detail, definition of what constitutes the scope of design process documentation, and the level of detail of Corrective Action Programs have changed in the industry; RNP has adopted changed procedures and processes.

In the late 1980s, an initiative was undertaken to reconstitute the design bases of the plant. This initiative is described in Attachment F.

In preparation of this response, RNP conducted a review of selected documents prepared between January 1, 1987, and December 31, 1996. Selection was made from the following:

1) Licensee Event Reports,
2) Notices of Violations,
3) Condition Reports,
4) NRC Inspection Reports,
5) Audit Reports by internal and external organizations, and
6) Internal Self Assessments.

As would be expected, there were instances of inadequate, incorrect, and omitted design bases requirements in appropriate procedures. In addition, there is evidence that was observed during the review, of RNP's initiatives to improve compliance with design bases requirements.

Examples include:

1) Reconstitution of the design bases described in Attachment F;
2) Improved administrative controls that provide instructions for preparation, revision, review, and approval of procedures;
3) Specific procedure improvement projects;
4) Initiation of an ongoing Self Assessment program; and
5) Ongoing conversion to improved Standard Technical Specifications, which requires a complete review and re-write of plant surveillance test procedures.

B. Procedure Development and Control

RNP Plant Administrative Procedures, upgraded during the improvement initiatives mentioned earlier, provide instructions for initiating, revising, and reviewing documents. Included are instructions for special procedures and infrequently performed tests or evolutions. Criteria are provided to control both assignment of review responsibility and scope of reviews, including review and approval of temporary changes to documents. Instructions are provided regarding the identification and revision of other documents that may be affected by the change being reviewed. In addition, directions involving the application of a 10CFR50.59 evaluation are provided, and a listing of individuals qualified to perform these reviews is in place.

Operations and Maintenance administrative controls governing each group's procedures, including test procedures, contain reference to the primary Plant Administrative Procedures for initiation, revision, and review instructions. In addition, Plant Modification and configuration control procedures contain guidance related to determining which documents may be affected by a proposed change.

In accordance with these procedures, individuals reviewing proposed changes to operating, maintenance, and testing procedures conduct evaluations consistent with 10CFR50.59; address other documents potentially affected by the proposed change, including relationships to design bases; and perform adequacy and accuracy reviews in accordance with a detailed scope of review instructions.

As described in Attachment A, the current design change process includes requirements for the identification and revision of operating, maintenance, testing, and other procedures affected by a modification. There are also programmatic links which identify required procedure changes when license amendments are implemented. Integral to both the modification process and the procedure change process is a requirement to review proposed changes consistent with the 10CFR50.59 program. Implementation of the 10CFR50.59 program is further discussed in Attachment A.

1. Accessibility and Retrievability

The accessibility and retrievability of design bases information for the procedure writer and procedure reviewers has been facilitated by the following tools:

1. Nuclear Revision Control System (NRCS) is an electronic mainframe system that provides current document revision level, status, outstanding changes, documents affected by change documents (e.g., ESRs), cross-reference, distribution profiles and history, and other document information. NRCS is a configuration management tool for identifying, verifying, providing status on cross-referencing, managing and

distributing design documents, change documents, plant operating manual documents, licensing documents, and other documents deemed essential to support the effective maintenance, operation, and configuration control of CP&L's nuclear plants;

2. Computer data base for retrieval of records (STAIRS);

3. Text electronic search capability as a reference for the UFSAR, Technical Specifications, Plant Operations Manual procedures, and industry operating experience.

RNP management believes the improved procedure development and control program provides reasonable assurance that design bases requirements have been adequately translated into appropriate procedures.

C. *Assessments and Reviews*

As stated above, in preparation for this response, RNP conducted a review of selected documents which were prepared between January 1, 1987, and December 31, 1996. Selection was made from the following:

1) Licensee Event Reports,
2) Notices of Violations,
3) Condition Reports,
4) NRC Inspection Reports,
5) Audit Reports by internal and external organizations, and
6) Internal Self Assessments.

The review was designed to identify conditions or trends in operating, maintenance, or testing procedures related to design bases requirement implementation. A general assessment of the results of this review was then performed to determine a level of confidence that plant design bases requirements have been translated into appropriate procedures.

Evidence of weaknesses in programmatic controls was noted in the documentation reviewed. Documented findings and corrective actions related to inadequate, incorrect, or omitted design bases requirements in procedures were observed. In addition, the documentation reviewed contained evidence that RNP began programmatic changes to address the adequacy of the design bases and the controls governing changes, in general. Improvement initiatives described below, the design bases reconstitution project described in Attachment F, and an increase in internal and external scrutiny such as implementation of the self-assessment program and NRC Augmented Inspection Team Audits, were products of recognition of the need for improvement.

These initiatives improved programmatic controls and technical accuracy of the targeted programs and strengthened linkage to design bases requirements.

D. *Improvement Initiatives*

1. 10CFR50.59 Evaluations

An effort to strengthen control of changes related to implementation of design bases requirements led to revisions of the RNP procedure that provides instructions for 10CFR50.59 evaluations of changes, tests, and experiments.

These revisions, adopted in 1990 and 1995, used NSAC-125 as guidance. The revisions incorporated detailed guidance on Unreviewed Safety Question (USQ) determinations including a step-by-step method for these determinations and a list of questions a reviewer should answer when making decisions. In addition, RNP developed and conducted a Qualified Safety Reviewer training program to instruct plant personnel regarding proper performance of safety evaluations.

Changes made to the 10CFR50.59 evaluation program represent a substantial improvement and have provided overall improvement in the quality of reviews compared to those performed previously. RNP management also believes this strengthened program provides added confidence that design bases requirements have been adequately translated into appropriate procedures.

2. Abnormal and Emergency Operating Procedures

RNP completed a program to upgrade Abnormal Operating Procedures (AOP) and Emergency Operating Procedures (EOP) in 1994. This effort was initially undertaken as a result of deficiencies identified in NRC Inspection Report 89-16 and continued in 1994 as a result of additional findings described in Inspection Report 94-07.

As a result of this project, program improvements include enhanced validation and verification that includes plant and simulator walk-through by organizations performing the procedure and inclusion of setpoints into a controlled calculation database, changes to which are required to be controlled by the document change program.

The EOP upgrade completed in 1994 was based on the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERG) Documentation and an RNP Plant Specific Technical Guidelines (PTSG) document. The RNP PTSG is comprised of the following: the WOG ERGs, ERG background documents, ERG Executive Volume, a Design Difference Document detailing the differences between the Westinghouse reference design on which the ERGs are based and the actual RNP design related to ERGs, a Generic Analysis Applicability Document, EOP Setpoint Document, Instrument Uncertainty Analysis, and the EOP Basis Document.

Collectively, these documents represent the design bases for EOPs. These bases are referenced in the operations procedure that provides instructions for EOP development. This EOP Program procedure also includes reference to the controlling Plant Administrative Procedure that provides instructions for document preparation, revision, and review.

These specific procedure upgrades and general programmatic improvements provide added confidence in the accuracy of AOPs and EOPs, as well as improved linkage to applicable design bases requirements.

3. Maintenance Procedures Upgrade

Findings from reviews of the RNP Maintenance Program, procedures, and experience during 1988 and 1989, indicated that improvement was necessary. Examples include:

1) The identification of lack of detail in procedures for some systems,
2) An inadequate maintenance procedure for a service water pump,
3) Recognition that hardware related problems contributed to the majority of forced outages, and
4) The occurrence of an increase in the number of NRC violations generally attributable to the maintenance/surveillance area during the evaluation period.

As a result, a program to review and improve maintenance procedures was implemented. The primary objective of the effort was to provide additional detail, technical accuracy, and focus attention to improving human factors in the new and revised procedures.

Cross-discipline review teams were organized which included, as appropriate:

1) Maintenance foremen and crew members,
2) Operations,
3) Health physics,
4) Work planning,
5) Quality control personnel, and
6) Systems engineers.

The program, which was completed in 1993, included assessments of the adequacy of procedure text, and where practicable, included a walk-through or actual performance of the procedure. More than 500 procedures were revised or created as a result of this effort.

The maintenance procedure upgrade program increased RNP confidence that procedures are technically accurate, contain the appropriate level of detailed instructions, and address human factors related to performance.

4. Technical Specification Review/Upgrade

During 1991 and 1992, RNP conducted an effort, in part, to provide added assurance that Surveillance Test Procedures that implement the testing requirements of Technical Specifications Table 4.1-1, "Maintenance Frequencies for Checks, Calibrations and Test of Instrument Channels," were technically correct. Eighteen discrepancies were identified during the project, two of which were reportable to the NRC and documented in Licensee Event Reports (LER) 92-002 and 92-010. The required changes to test procedures were identified and have been completed. Certain previously accepted testing methodology, which was based on interpretations of applicable Technical Specifications, was determined to be incorrect in 1994. This prompted a supplemental review of Section 4 of the Technical Specifications. Additional discrepancies were identified and addressed.

Problems in interpretations of the RNP custom Technical Specifications, such as described above, led to the conclusion that adoption of Improved Technical Specifications (ITS) was appropriate. Accordingly, RNP developed and submitted a license amendment package on August 27, 1996, to convert RNP to the ITS. Development of this package required extensive technical reviews of selected plant design documents. Currently, the expected approval date is June 1997.

During the process of conversion to ITS, a comparison review of RNP current Technical Specification Table 4.1-1 with counterpart ITS Tables 3.3-1 and 3.3-2 criteria was completed. Existing test procedures are being revised and new procedures prepared as appropriate to comply with ITS surveillance test requirements.

These initiatives improved the technical quality of the testing procedures and provided linkage to design bases requirements.

E. Conclusion

The rationale for concluding with reasonable assurance that design bases requirements have been adequately translated into operating, maintenance, and testing procedures is based on the following key elements:

1) Processes for procedure development and control are established and implemented;
2) The Design Bases Document reconstitution project improved the availability and retrievability of design documents;
3) The RNP 10CFR50.59 program has been revised and improved. This has been recognized by the NRC in IR 91-05 as having resulted in improvement in the quality of safety reviews;
4) Processes for design changes which include programmatic links to procedures are established and implemented;
5) A program of periodic self-assessments is established and implemented; and

6) The RNP Corrective Action Program that requires documentation of discrepancies and tracking of corrective actions to completion is established and functioning.

The elements listed above provide confidence that RNP procedures are generally consistent with the design bases. Deficiencies have occasionally been identified in operating, maintenance, and testing procedures. However, when their number, type, safety significance, and resulting corrective action are considered in the whole, these deficiencies do not change our conclusion today. CP&L will continue to assess processes and procedures and make changes as necessary for continual improvement.

Attachment C
Consistency of SSC Configuration and Performance with Design Bases

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I. Introduction

This attachment provides information in response to NRC Item (c):

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

CP&L's conclusions have been based upon plant-specific reviews of both the current administrative control processes that govern the control of system, structure, and component configuration and of the current testing and monitoring programs that demonstrate the performance of the systems, structures, and components. Relative to the design bases, these reviews considered the scope and content of these processes and programs and their evolution as additional requirements were instituted to resolve identified deficiencies. These processes and programs are subject to reviews and assessments. Thus, they are changed from time-to-time to strengthen or provide adjustments, as required, to reflect improvements or organizational changes. Each plant-specific section in this attachment is organized into six subsections:

A. **Summary** – an overview of process by which the rationale was derived;
B. **SSC Configuration** – administrative control processes for SSC configuration;
C. **SSC Performance** – testing and monitoring programs for SSC performance;
D. **Assessments and Reviews** – the results of pertinent internal and external evaluations;
E. **Improvement Initiatives** – descriptions of relevant configuration control improvements;
F. **Conclusion** – rationale for concluding that system, structure, and component configuration and performance are consistent with the design bases.

Where appropriate, ongoing efforts have been undertaken to strengthen these processes and programs. These efforts are listed in Attachment G as either commitments or enhancements.

II. Brunswick Nuclear Plant (BNP)

A. *Summary*

Each Brunswick nuclear unit is a General Electric (GE) Boiling Water Reactor (BWR) with a "Mark I" reactor containment design. Unit 2 received its operating license in December 1974, followed by Unit 1 in September 1976. GE performed design activities for the Nuclear Steam Supply System (NSSS), and United Engineers and Constructors, Inc. (UE&C) was the architect/engineer for Balance of Plant (BOP) systems, structures, and components. The initial Structure, System, and Component (SSC) configurations were established based primarily on original GE & UE&C design drawings and calculations, the Final Safety Analysis Report (FSAR), and the Nuclear Regulatory Commission (NRC) Safety Evaluation Reports (SER). Plant documentation, such as drawings and calculations for BOP systems, structures, and components, was initially controlled by UE&C and GE off-site.[1]

As discussed in Attachment A, programs and processes are in place so that, when design changes to the plant are made, the configuration and performance of SSCs remain consistent with design bases requirements. Other configuration control measures are discussed below.

There are a number of performance monitoring and testing programs in place at BNP that periodically confirm that SSC performance is within specified acceptance limits. These monitoring and testing programs provide confidence that the SSCs are capable of performing their intended safety functions in accordance with the design bases requirements. These monitoring and testing programs are also described below.

The sections below also describe a number of self-assessments and focused review efforts that have assessed the consistency of the configuration and performance of SSCs with their design bases requirements. These reviews concluded that the configuration and performance of SSCs have generally remained consistent with the design bases. These efforts have, however, resulted in the identification of some discrepancies related to the configuration or performance of SSCs. CP&L takes such findings seriously and has responded to the issues by determining the extent and significance of the deficiencies and taking appropriate corrective actions. CP&L's Corrective Action Program is described further in Attachment D.

[1] GE retains most design engineering information for the NSSS systems as proprietary with non-proprietary design information and plant drawings controlled by CP&L.

Based on the above, CP&L concludes with reasonable assurance that, notwithstanding previously identified deficiencies, the configuration and performance of SSCs are generally consistent with design bases requirements. The following sections provide additional detail to substantiate CP&L's rationale. CP&L has an ongoing self-assessment program to increase confidence that the configuration and performance of SSCs at BNP are consistent with the design bases. These actions are discussed in Attachment G.

B. Structure, System, and Component Configuration

A number of programs, processes, and procedures are used to govern the control of SSC configuration at BNP. These include the:

1) Engineering design control processes described in Attachment A;
2) Design bases documents described in Attachment F;
3) Plant Equipment Data Base System (EDBS); and
4) Various procedures that control the operation and maintenance of SSCs.

These items are described below.

1. Design Control Process

The controls over plant modifications are described in Attachment A. Significant process improvements were developed and implemented (e.g., Engineering Service Requests (ESR) and Corrective Action Program) during and following the 1992-1993 dual unit shutdown. As part of the basis for restart, CP&L implemented its Three-Year Plan initiative, Integrated Action Plan, and the NGG common plant procedures initiative. Beginning with implementation of the ESR process (originally per procedure PLP-30, now procedure EGR-NGGC-0005) in 1994, engineering backlogs were significantly reduced. The reduction of these backlogs contributed significantly to the ability to track in realtime, current SSC configurations.

BNP core reloads are managed by CP&L using the ESR process. The fuel vendor (GE) generally performs the analyses of record that are used to confirm that the reload meets the appropriate design goals, using NRC approved methodology. CP&L performs an extensive review of the GE documents. These reviews include detailed owner reviews and may include independent calculations to confirm the results of the GE analyses.

Once the core has been loaded and verified through a series of independent checks, the ESR package is closed. At this point, the normal plant procedures governing startup testing are followed to verify that the core is operating as expected. These procedures include tests performed from criticality (shutdown

margin) to full power (reactivity anomaly and predicted to measured comparisons).

2. Equipment Data Base System (EDBS)

The EDBS is a computerized database of plant components organized according to equipment identification or tag number. Information contained in EDBS includes verified and controlled information such as component quality class, environmental qualification, engineering design and reference data, PM route maintenance, Appendix R data, setpoints, Regulatory-Related Instrument List (RRIL) data, and, in some cases, field verification of information and validation against approved design documents. EDBS is used in planning equipment repairs, in preparation of equipment clearances and in engineering analysis and design work. Changes to EDBS are procedurally controlled so that appropriate reviews and approvals are performed. The EDBS is used by plant personnel involved with engineering, maintenance, and operation of BNP. The use of EDBS described above provides additional assurance that the plant configuration is maintained consistent with design requirements.

3. Operational Configuration Control

Requirements are in place to provide for the control of plant component configuration during plant operation. As described in Attachment B, plant operating and testing procedures receive extensive review and approval. Operational configuration controls include:

1) Valve line-up sheets;
2) Electrical line-up sheets;
3) Selective independent verification of component positioning;
4) Locking of selected valves; and
5) Use of equipment clearances.

4. Operator Workarounds

Conditions that affect the operator's ability to operate plant equipment are identified and tracked by the Operator Workaround program. This program provides increased management attention and scheduling priority for maintenance work requests and ESRs associated with these conditions. The program includes provisions to identify design deficiencies which prevent operation of a system as originally designed.

5. Short-Term Structural Integrity

During the 1980s, a large backlog of discrepant structural conditions accumulated at BNP. Specifically, these were conditions that satisfied short-term structural integrity criteria (STSI), but did not meet all applicable code requirements. In the early 1990s, a focused effort to reduce engineering backlogs resulted in the resolution of long-standing STSI items. As defined in the UFSAR, STSI limits are primarily used to evaluate the acceptability of "as-found" conditions in order to justify continued plant operations. STSI conditions are permitted to remain until the next refueling outage at which time they are required to be resolved.

C. *Structure, System, and Component Performance*

A number of testing and monitoring programs are used to measure the performance of SSCs at BNP. These include:

1) Technical specification surveillance testing;
2) Inservice testing and inservice inspection;
3) Performance monitoring in accordance with the NRC Maintenance Rule (10CFR50.65);
4) Post-maintenance and post-modification testing; and
5) Post-trip reviews.

These testing programs help to confirm that the performance of SSCs remain within established limits. The specific programs described in the following sections provide additional bases for reasonable confidence that SSCs are performing their intended safety functions in accordance with design bases.

1. Technical Specification Surveillance Testing

Technical Specification surveillance testing is a principal method to show that SSC performance is consistent with design bases parameters. The Technical Specification surveillance testing program provides for periodic testing of selected SSCs such as pumps, valves, and instrumentation to demonstrate that their actual performance is consistent with the specified acceptance limits. This surveillance testing shows, to the extent practicable, that the SSCs are capable of performing their intended safety functions. For example, ECCS pumps are periodically tested to demonstrate that developed head and flowrate meet design requirements.

The Bases Section of the Technical Specifications frequently contains, or references the source of, specific design bases information to support the acceptance limits for surveillance tests. BNP has recently reviewed the content of the Technical Specification Bases related to surveillance testing requirements as a

part of BNP's conversion to Improved Technical Specifications (ITS).[2] In support of this effort, information in the Bases Section for surveillance testing requirements included in the ITS was verified. Where necessary, calculations were reconstituted to confirm that the specified acceptance limits were adequate.

2. Inservice Testing and Inservice Inspection

Inservice testing and inspections performed to comply with ASME Section XI also provide confidence that SSC configuration and performance are consistent with design. This program covers a wide range of safety-related components including pumps, valves, piping, and supports. It is intended to detect service-related performance degradation. If testing indicates performance deficiencies, testing frequency is increased and/or equipment is declared inoperable and corrective actions are taken.

3. Maintenance Rule

Performance of systems that support safe plant operation is tracked in accordance with the Maintenance Rule per 10CFR50.65. Essential elements of this program include:

1) Establishment and application of risk-significant criteria;
2) Setting system, structure, and component performance criteria;
3) Trending system performance to demonstrate the effectiveness of maintenance activities;
4) Setting system performance goals; and
5) Periodically assessing performance.

The program contains requirements to perform root cause evaluations and implement corrective actions when a system fails to meet established performance criteria.

[2] By letter dated November 1, 1996, CP&L proposed amendments to the Brunswick Plant Technical Specifications to upgrade the format and content to be consistent with NUREG-1433, Rev. 1, "Standard Technical Specifications for General Electric Plants, BWR4."

4. Preventative and Predictive Maintenance Programs

There is a preventative maintenance program which provides additional assurance that components will function properly when called upon. There is also a predictive maintenance program to provide early detection and diagnosis of equipment problems or deterioration allowing appropriate corrective action prior to failure.

5. Post-Maintenance and Post-Modification Testing

Following preventative or corrective maintenance of plant equipment, post-maintenance testing is performed to verify that equipment function and performance are restored. At BNP, post-maintenance testing is controlled in accordance with procedure 0PLP-20 which provides the necessary controls and guidelines for the selection, performance, and documentation of required post-maintenance testing based on the maintenance activity performed. Acceptance testing is also performed after SSC modifications to compare SSC performance with specified acceptance criteria, defined by engineering, to provide assurance that the modified SSC complies with design bases requirements. Post-modification testing is described in Attachment A.

6. Post-Trip Reviews

The plant procedures require a review of plant equipment performance following each plant trip. This provides additional opportunities to detect equipment performance issues that may be indicative of operation outside the design bases.

7. Additional Testing

Periodic performance testing is also performed to meet specific reliability or programmatic requirements. Examples include piping inspections for flow accelerated corrosion, fire protection component testing, and MOV testing. The methods and programs described above provide additional confidence that plant performance continues to meet applicable design requirements.

8. Additional Programs

Additional programs exist which help to provide assurance that plant configuration is maintained within design bases. Examples include self-checking initiatives such as STAR, internal and external Operating Experience Feedback (OEF), interactions with the BWR Owners' Group, Quality Check, and employee suggestions.

D. Assessments and Reviews

As described in Attachment B, to confirm the adequacy of configuration control and performance monitoring programs, CP&L has performed periodic in-depth assessments for selected systems and programs at BNP such as the SWOPI, HPCI SSFI, and CBEAF SSFI.

In response to an error that was discovered in the Unit 2 process computer heat balance calculation program, other computer applications were reviewed. Evaluations were performed to confirm that proper controls were established during software development, installation, and testing to provide assurance of the operational functionality of the systems.

In addition, a broad-based plant startup readiness assessment and startup/power ascension program was performed in 1993.

An Electrical Distribution System Functional Inspection (EDSFI) conducted by the NRC in 1991 provided another avenue for BNP to identify weaknesses in plant systems. This inspection, its findings, and subsequent follow-up, are also discussed in Attachment B.

In association with the review of the power uprate analyses, questions were raised regarding the calculations of post-accident doses to control room operators. The resolution of these questions is under evaluation at this time.

These assessments have demonstrated the adequacy of the target systems or programs, as well as provided a level of confidence in the effectiveness of the governing procedures and processes.

As of June 14, 1996, a review identified programmatic weaknesses when multiple examples of where the Environment Qualification Program, required per 10CFR50.49, had failed both to maintain a complete list of equipment important to safety and to maintain auditable documents of the equipment qualification. This resulted in the NRC Notice of Violation on November 19, 1996, with the basis provided in Inspection Report 96-14. Examples also showed where environmental profiles did not reflect the most severe design bases accident or had

not been updated for changed radiation conditions that resulted from the Hydrogen Water Chemistry modifications. Examples were found of items that had been removed from the program without documenting appropriate justification or receiving a management review. Reviews and corrective actions to evaluate and strengthen the Environment Qualification Program are in progress. BNP has completed review and validation of the BNP EQ Master List as of December 1996, and reevaluation of the environmental design conditions for each plant area and event. An effort is ongoing to enhance the existing qualification data packages for equipment on this list. The CP&L letter BSEP 96-00476 dated December 19, 1996, contains a description of planned and completed actions in this area.

E. Improvement Initiatives

Since the time of the original licensing of the plant, BNP has initiated various projects that presented opportunities to compare the as-built configuration of the plant with design bases documents. BNP initiated such projects either as part of plant modifications and enhancements, or as corrective actions to address deficiencies identified by BNP, the NRC, or through other industry experience.

1. Design Basis Reconstitution

In the past few years, BNP has reconstituted a substantial portion of the plant design bases through several discrete efforts. These efforts include the:

1) DBD Reconstitution;
2) Design Turnover Project (DTOP);
3) Electrical Calculation Reconstitution;
4) Miscellaneous Steel Verification; and
5) 24-month Fuel Cycle Setpoint Reconstitution.

2. Restart Readiness from Dual Unit Shutdown

Following the April 21, 1992 forced outage of the Brunswick Nuclear Plant (BNP) Units 1 & 2, procedural guidance was developed for a line-management assessment and Plant Nuclear Safety Committee (PNSC) determination of the readiness to safely and reliably startup and operate the units through the next operating cycle. The process was described in Plant Notices PN-30, "Integrated Recovery Methodology" and PN-31, "Line-Management Self-Assessment of Readiness for Restart of BNP Unit 2" (at Rev. 2), and "Systems Turnover to Operations and Line-Management Self-Assessment of Readiness for restart of Unit 1" (at Rev. 3). This process assessed the status of each plant system and organization. It was performed in 1992 and 1993 for both units to support plant startups.

The system readiness assessment included a structured plan for systems turnover and a determination of readiness by the respective system engineer affirming that each system was ready to support safe and reliable startup and power operation through the next operating cycle. This review was based upon:

1) Current system conditions that could have changed since the shutdown of the plant, as verified by walk-down;
2) Completion of identified startup required actions; and
3) An assessment of backlogged work items.

As part of this readiness assessment/affirmation, system walk-downs were performed by the responsible system engineers.

The organizational readiness assessment provided a structured methodology to show that each selected organization was in a state of readiness to support safe and reliable startup and power operation through the next operating cycle.

Organizational readiness was affirmed by the responsible organization managers that necessary startup actions including regulatory commitments had been completed. Based on the results of these affirmations, the Plant General Manager made a recommendation to the Director of Site Operations and the Vice President - BNP regarding the readiness to restart and operate the units.

After the determination was made by the Director of Site Operations to commence startup, an integrated startup and power ascension test program was implemented in Special procedures 2SP 93-019, "Unit 2 Startup and Power Ascension Guidelines and Checklists," and 1SP 93-058, "Unit 1 Startup and Power Ascension Guidelines and Checklists." These procedures established the methodology for the safe, controlled, and deliberate return to service of the Brunswick Units. These procedures provided guidance and checklists for tests, inspection activities, and hold points required during plant startup and power ascension. They included the requirement to perform comprehensive testing of plant systems for reliability of service and compliance with plant design bases and technical specifications. In addition, walk-downs of selected plant systems were performed by assigned Operations and Systems Engineering personnel. These walk-downs focused on the areas of material condition, radiological protection, and housekeeping. Hold points were established at predetermined power levels to assess plant performance.

This process assessed the status of each plant system and site organization, and provided further assurance that the Brunswick units were operating within their design bases.

3. Power Uprate

This project included a review of the impact of power uprate on:

1) Reactor Core and Fuel Performance;
2) Reactor Coolant System and Connected Systems;
3) Engineered Safety Features;
4) Containment System Performance;
5) Power Distribution Systems; and
6) Reactor Pressure Vessel and Internals.

These areas were reviewed by General Electric and CP&L. As a result, numerous design calculations were reviewed, verified, reconstituted, or performed, especially in the area of Emergency Core Cooling Systems (ECCS) performance and Technical Specification instrumentation.

This project presented many opportunities to review the plant design bases and to identify and correct existing discrepancies. For example, during the implementation stage of Power Uprate (PUR) on Unit 1, an error was identified in the Station Blackout analysis during the calculation update. It was determined that the SBO analysis that relied on this calculation did not identify that the calculation contained inputs and acceptance criteria that were incorrect. Therefore, a review was performed of the input assumptions of calculations that were used for PUR. To provide assurance that calculations affected by PUR were identified, CP&L performed a review of the calculations of record for Unit 1 and those common to both units. Based on this review, it has been concluded the calculations and analyses used for PUR are correct.

4. Hot-side/Cold-side Walk-downs

The NRC raised issues involving the material condition of the plant in Inspection Reports 50-325/92-12 and 50-324/92-12. The BNP reply to these issues in letter NLS 92-160, dated July 23, 1992, committed to area cold-side (radiologically accessible) and hot-side (radiologically not accessible until conditions permit access) walk-downs. The walk-downs were performed following the dual unit shutdown. A formal procedure was not used in performance of the original hot-side/cold-side walk-downs. General guidance and a pre-job briefing were provided for the walk-down team members prior to starting the walk-downs. Walk-down team members had varying levels of qualification and experience, and the consistency of the walk-downs varied. For this reason, Brunswick management decided to perform a comprehensive inspection of plant spaces for material condition against higher management standards. Procedure 0SP-92-076 provided the standards, which included the recommendation of the INPO Good Practice for Plant Inspection Programs, and was approved on November 19, 1992.

Per this procedure, walk-downs were performed by a multidisciplined team of individuals comprised of mechanical, electrical/I&C, and civil structural skills and knowledge. Personnel were trained on how to identify deficiencies, and inspections were performed under the supervision of persons trained in INPO observation techniques.

As part of this effort, an outside consultant (EQE Engineering Consultants, or EQE) performed a review of the anchorage and structural support considerations for floor-mounted, safety-related electrical equipment at Brunswick. This review was performed to address issues identified by plant QA personnel that equipment may not have been installed in accordance with plant design drawings. A total of 154 components were evaluated including the DC switchboards in the battery rooms and the 4160 volt switchgear in the Diesel Generator Building.

Walk-downs of the equipment were performed with comparison of the anchorage to plant documentation. Results of this review determined that, while many panels and cabinets deviated from the original design drawings, the existing anchorages to the embedded channels were capable of carrying the required loads with no operability issue identified.

The confirmation of plant material condition provided added assurance regarding the reliability of SSCs to satisfy performance requirements consistent with design parameters.

5. Generic Letter 88-20 -- Individual Plant Examination of External Events (IPEEE)

CP&L issued a report in June 1995 documenting completion of the IPEEE requirement per Generic Letter 88-20. The effort was conducted in accordance with NUREG 1407. Completion of the IPEEE effort involved cognizant engineers performing field walk-downs in support of the final report. Several walk-downs were performed in the seismic area including inspection of equipment anchorages included in the chosen safe shutdown path. When existing configurations and conditions were found to be outside of the acceptance criteria set forth in NUREG 1407 and CP&L guidelines for the walk-downs, they were resolved via:

1) Housekeeping;
2) Maintenance, repair or modification;
3) Further investigation; or
4) Demonstration through High Confidence of Low Probability of Failure (HCLPF) analyses.

Confirmation of equipment configuration, and evaluation and resolution of identified discrepancies provide additional assurance that the SSCs reviewed will perform their design bases functions.

6. Unresolved Safety Issues (USI) A-46

Report titled "Brunswick Nuclear Plant USI A-46 Seismic Evaluation Report" provided documentation of the USI A-46 effort at Brunswick with corrective actions to resolve program "outliers" (i.e. existing conditions or configurations not generically evaluated and qualified). Evaluations performed in support of the A-46 report were based on the Generic Implementation Procedure (GIP) developed by the Seismic Qualification Utility Group (SQUG).

Detailed plant walk-downs were performed for the identified equipment on the selected safe shutdown equipment list (SSEL). Walk-downs were performed in accordance with the SQUG GIP and enhancements based upon guidance contained in the Electric Power Research Institute document number NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin."

No significant or programmatic deviation from the SQUG GIP was required for the Brunswick A-46 program. Outliers, items not meeting the evaluation criteria as set forth in the GIP and requiring further analytical evaluation, are scheduled for resolution by the end of the Unit 1 refueling outage scheduled for the spring of 1998.

Together the above projects presented many opportunities to identify and correct specific or programmatic deficiencies in the configuration or performance of SSCs. Therefore, these efforts provide additional assurance that the configuration and performance of SSCs is consistent with design bases documents.

F. Conclusion

The rationale for concluding with reasonable assurance that plant design bases have been adequately incorporated into system, structure, or component configuration and performance is based on the following key elements:

1) Established programs to maintain the design bases and configuration control;
2) Successful restart testing results from the 1992 extended outage;
3) Comprehensive surveillance testing program;
4) Established preventative and predictive maintenance programs;
5) Established post-maintenance and post-modification testing programs; and
6) An established Corrective Action Program.

Deficiencies have been periodically identified. However, when their number, type, safety significance, and resulting corrective actions are considered in the whole, these deficiencies do not change our conclusion today. CP&L will continue to assess SSC configuration and performance and to address identified deficiencies in accordance with the Corrective Action Program.

III. Harris Nuclear Plant (HNP)

A. Summary

The Harris Nuclear Plant is a pressurized water reactor with a licensed power rating of 2775 MWt. A full-power operating license was issued to HNP in January 1987. The Nuclear Steam Supply systems were designed and supplied by Westinghouse. The architect engineer for Balance of Plant (BOP) systems, structures, and components of record was Ebasco. The initial SSC configurations were established based primarily on original Westinghouse & Ebasco design drawings and calculations, the Final Safety Analysis Report (FSAR), and the Nuclear Regulatory Commission (NRC) Safety Evaluation Reports (SER).

Prior to initial license receipt, activities, as described in Section III.B.1 below, were performed which resulted in a verification that the Harris Nuclear Plant was designed, constructed, and tested in accordance with representations and descriptions contained in the Final Safety Analysis Report and supporting documents. Programs were in place at the time of initial license receipt to help maintain the design bases and to help verify configuration control. These programs have been strengthened as experience has been gained. The current program is described in Attachment A.

Pre-operational and startup testing provided confidence that the performance of SSCs met the acceptance criteria established. Since startup, a number of performance monitoring and testing programs have been in place with the objective of periodically confirming that the performance of various SSCs is within specified acceptance limits. These monitoring and testing programs, in part, provide confidence that SSCs are capable of performing their intended safety functions in accordance with the design bases requirements. These monitoring and testing programs are described in this attachment.

To enhance confidence that these programs are effective in maintaining SSC configuration and performance consistent with the design bases, internal and external assessments are conducted. Summaries of some of these assessments are included in this response.

When discrepancies were identified, CP&L corrected the exception and took action, as appropriate, to prevent recurrence. The following response provides additional detail to substantiate CP&L's rationale. In certain areas, CP&L has determined that additional reviews or actions are appropriate to provide additional assurance that the configuration and performance of SSCs at HNP are consistent with the design bases. These actions are discussed in Attachment G.

B. Structure, System, and Component Configuration

1. Consistency at the Time of Plant Licensing

Prior to initial licensing of HNP, several reviews and evaluations were performed to verify consistency among the Final Safety Analysis Report (FSAR), proposed Technical Specifications, and design bases documents.

At CP&L's request, the plant's Architect/Engineer firm, Ebasco, performed a consistency review of the FSAR. The stated purpose of the review was to provide a documented source of verification for consistency within sections of the FSAR, consistency with related licensing and design documents, and consistency with commitments made by CP&L.

The accuracy of the HNP Technical Specifications was to be established through an extensive and integrated development process. The development process included reviews by CP&L personnel, Ebasco, and Westinghouse. A matrix of Technical Specification values and accident analysis assumptions was developed. Westinghouse provided assurance to CP&L that reasonable steps to verify the accuracy and completeness of the inputs had been taken.

Ebasco conducted a Technical Specification consistency review and a final interdisciplinary review to verify the accuracy and completeness of the proposed "final" HNP Technical Specifications relative to the inputs provided by Ebasco. A cross reference between the FSAR and Technical Specifications was developed. Ebasco certified the accuracy of the proposed "final" Technical Specifications.

The proposed "final" HNP Technical Specifications also received a multi-disciplinary review within CP&L. This included reviews by engineering, regulatory, and quality assurance organizations. The FSAR, Safety Evaluation Report (SER), and docket file commitments were reviewed for impact on the Technical Specifications.

The Design Basis Document development for HNP is discussed in Attachment F.

On October 3, 1986, CP&L notified the NRC that the HNP and its operating organization would be ready to receive an operating license on October 10, 1986. As part of the notification, credit was taken for the tracking systems used to track completion of outstanding construction items, required hardware modifications, and test exceptions identified during pre-operational testing and inspections. A similar certification affirming the accuracy of the HNP Technical Specifications was submitted on October 6, 1986.

These activities were performed to provide additional confidence that systems, structures, and components configuration were consistent with the design bases at the time of initial plant licensing.

2. Operational Configuration Control

Subsequent to licensing, CP&L has relied on programs described in Attachment A and their predecessors to maintain the design bases. In addition to those described in Attachment A, a number of programs, processes and procedures are used to govern the control of SSC configuration at HNP. These include the:

1) Plant Equipment Data Base System (EDBS);
2) Core Reload Controls; and
3) Configuration controls for operation and maintenance of SSCs.

Equipment Data Base System (EDBS)

The EDBS is a computerized database of plant components organized according to equipment identification or tag number. Information contained in EDBS includes information such as component quality class, environmental qualification, and engineering design and reference data. EDBS is updated as a result of changes made under the ESR process.

Core Reload Controls

HNP core reloads will be controlled using the ESR process. The ESR process provides for the identification of impacts to plant programs and organizations and for required reviews. In addition, the ESR process requires that affected plant documents and databases be identified.

The ESR for the next core reload involves providing the core design, analyses, data and fuel to support operation. This includes interface with the vendor. By procedure, a review is performed on designs and/or design output documents received from external sources (such as a fuel vendor.) This review is to determine the acceptability and validity of the externally generated document.

Following reactor core refueling, a procedure provides the necessary instructions for a verification of correct fuel loading. Procedures also provide the methodology for a safe, well-controlled, and deliberate plant startup and power ascension following the refueling. One of the performance objectives of the power ascension testing program is to conduct the appropriate tests to verify equipment needed for safe and reliable plant operation performs within required limits.

Configuration Controls for Operation and Maintenance of SSCs

Requirements are in place for control of plant component configuration during plant operation. As described in Attachment B, plant operating and testing procedures are to be reviewed and approved. Operational configuration controls include:

1) Valve line-up sheets;
2) Electrical line-up sheets;
3) Independent verification of component positioning;
4) Locking of selected valves;
5) Use of equipment clearances;
6) Control of on-line maintenance; and
7) Tracking systems for inoperable and out-of-normal position equipment.

Conditions that affect the operator's ability to operate plant equipment are identified and tracked by the Operator Workaround program. This program provides increased management attention and scheduling priority for maintenance work requests and ESRs associated with these conditions. The program includes provisions to identify design deficiencies which prevent operation of a system as originally designed.

These controls provide additional confidence that component configuration has been and will be maintained consistent with design bases requirements.

C. *Structure, System, and Component Performance*

NRC item (c) also requested information regarding the rationale used for concluding that system, structure, and component performance is consistent with the design bases. Pre-operational testing and testing programs implemented since initial plant operation contribute to the confidence that SSCs are capable of performing their design bases functions. These programs are periodically assessed by the line organization and independent parties. Through these assessments, program improvements or enhancements are identified. Also, if program deficiencies are identified, corrective actions are implemented through the Corrective Action Program described in Attachment D.

Summaries of the key attributes of pre-operational testing and of operational testing and equipment monitoring programs which provide this confidence are described in this attachment.

1. Startup Testing

The HNP Power Ascension Testing Program included single and multisystem tests commencing with initial fuel loading and continuing through full power. These tests were conducted to demonstrate overall plant performance and included such activities as cold and hot shutdown testing, zero power tests, and power ascension tests. Acceptance criteria were established and compared to test results. The test results were submitted to the NRC on July 30, 1987.

2. Technical Specification Surveillance Testing

Technical Specification surveillance testing is a principal method of confirming that SSC performance is consistent with design bases parameters. The Technical Specification surveillance testing program provides for periodic testing of selected SSCs such as pumps, valves, and instrumentation to confirm that their actual performance is consistent with the specified acceptance limits. The Bases Sections of the Technical Specifications contain or reference specific design bases information to support the acceptance limits for surveillance tests. This surveillance testing confirms, to the extent practicable, that the SSCs are capable of performing their intended safety functions. For example, ECCS pumps are periodically tested to demonstrate that developed head and flowrate meet design requirements.

HNP uses an electronic scheduling system to ensure that surveillances are conducted at the frequencies required by the Technical Specifications. If unacceptable results are obtained, HNP Technical Specifications require specific actions to be taken. If necessary, the SSC is declared inoperable and corrective actions are taken. After corrective measures are taken, testing is required to be repeated to assure acceptable performance.

As described in Attachment B, some deficiencies have been identified in the surveillance testing program. HNP plans to complete the Surveillance Program Review Project described in Attachment B.

3. Inservice Testing and Inservice Inspection

Inservice testing and inservice inspections performed to comply with ASME Section XI also provide additional confidence that SSC configuration and performance are consistent with design. This program covers a wide range of safety-related components including pumps, valves, piping, and supports. It is intended to detect service-related performance degradation. If testing indicates performance deficiencies, the program provides for an increase in testing frequency, for a declaration of equipment inoperability, and implementing corrective actions.

4. Maintenance Rule

Design-related performance of systems that support safe plant operation is tracked in accordance with the Maintenance Rule per 10CFR50.65. Essential elements of this program include:

1) Establishment and application of risk-significant criteria;
2) Establishing SSC performance criteria;
3) Trending system performance criteria; and
4) Periodically assessing performance.

The program contains requirements to perform evaluations and implement corrective actions when a system fails to meet established performance criteria.

5. Preventative and Predictive Maintenance Programs

Preventative and predictive maintenance programs are in place to help support reliable equipment performance. The preventative maintenance program provides additional assurance that components will function properly when called upon. The predictive maintenance program provides early detection and diagnosis of equipment problems or deterioration allowing appropriate corrective action prior to failure. Following preventative maintenance, post-maintenance testing is to be performed to show correct component performance.

6. Post-Maintenance and Post-Modification Testing

Following preventative or corrective maintenance of plant equipment, post-maintenance testing is to be performed to demonstrate that equipment function and performance are restored. Post-maintenance testing is controlled in accordance with procedures which provide the necessary controls and guidelines for the selection, performance, and documentation of required post-maintenance testing based on the maintenance activity performed. Acceptance testing is also performed after SSC modifications to compare performance with design bases requirements. Post-modification testing is described in Attachment A.

7. Post-Trip Reviews

The post-trip review procedure requires a review of plant equipment performance following each plant trip. This review provides additional opportunities to detect equipment performance issues that may be indicative of operation outside the design bases. Procedures require as part of the post-trip review that the plant response be compared to corresponding transient response described in the FSAR in an effort to identify abnormal or degraded performance. Forty five (45) post-trip reports, for the period from December 12, 1986 to January 31, 1997, were

reviewed to determine if there were recurring problems with protection or safeguards equipment. No malfunction of protection or safeguards equipment was noted in 33 of the 45 reactor trips.

The only recurring equipment problem involved the Turbine Driven Auxiliary Feedwater (TDAFW) Pump. The pump tripped due to electrical overspeed conditions in three reactor trip events prior to October 9, 1989. The problems with TDAFW were identified and resolved as evidenced by improved performance on demand. Since October 9, 1989, Auxiliary Feedwater (AFW) has performed satisfactorily in nine (9) of nine (9) automatic actuations following reactor trips.

Other isolated equipment problems, including one failure of a reactor trip breaker to automatically open, have been identified during these post-trip reviews. The restart authorization requires consideration of whether:

1) The major safety-related and other important equipment functioned properly during the transient, or
2) Corrective action and satisfactory testing have been performed or will be completed when plant conditions permit.

8. Additional Testing

Additional performance testing is also performed to meet specific reliability or programmatic requirements. Examples include:

1) Piping inspections for flow accelerated corrosion;
2) Piping and heat exchanger inspections for biological fouling;
3) Fire protection component testing;
4) Thermography;
5) Lube oil sampling; and
6) MOV testing.

Due to the extensive equipment performance monitoring, analysis, and repair programs at HNP, there is additional confidence that issues regarding equipment performance relative to design bases are promptly identified and resolved.

D. *Assessments and Reviews*

In preparation for this response, the following were reviewed for the time period between October 24, 1986 and December 31, 1996 to identify any significant programmatic weakness in the configuration control processes which could affect safe plant operation:

1) Licensee Event Reports;
2) NRC Notices of Violation;
3) Inspection Reports;
4) Self-assessments; and
5) Independent internal assessments (Quality Assurance, Nuclear Assessment Section, and Performance Evaluation Section reviews).

Recent events were also reviewed to determine the nature and extent of existing weaknesses. A summary of the review and a description of recent significant issues are provided below.

1. Licensee Event Reports (LER)

LER summaries were reviewed to identify reportable conditions or events directly associated with system, structure, or component configuration. Twelve (12) LERs have been reported under regulation 10CFR50.73(a)(2)(ii), unanalyzed condition or operation outside design bases. Two of these reports were a result of component failure, two resulted from personnel error, one from an inadequate procedure, and seven from design problems. The LERs were further reviewed to determine whether there was a pattern involving specific SSCs or programs. The review did not identify a common cause related to SSC configuration control. Corrective actions for these reportable events are dispositioned through the Corrective Action Program described in Attachment D.

2. NRC Notices of Violation

NRC cited and non-cited violations received since initial licensing were reviewed. Notices of Violations (NOVs) involving configuration or design were identified. Thirteen (13) violations of 10CFR50 Appendix B Criterion III (Design Control) and thirteen violations of 10CFR50 Appendix B Criterion V (Instructions, Procedures, and Drawings) have been received. (Note that two of the 13 violations of Criterion III are included in the LERs described above.) One of the violations of Criterion III was determined to be a Level III violation. This Level III violation is discussed in the following section regarding service water system configuration control.

The NOVs were further reviewed to determine whether there was a pattern involving specific SSCs or programs. The review did not identify a common cause related to SSC configuration control. Corrective actions for violations are dispositioned through the Corrective Action Program described in Attachment D.

3. Major Self-Assessments and Independent Assessments

Self-assessments and assessments performed by Quality Assurance, Nuclear Assessment Section, and Performance Evaluation Section involving design were identified and reviewed for indications of a programmatic deficiency in configuration control. Although several assessments over the years have identified issues and/or weaknesses in the configuration control process, in general, CP&L believes the programs to be effective. The significance of these issues and resultant corrective actions from the assessments are discussed below.

Electrical Distribution System Functional Inspection

One of the most extensive NRC inspections at HNP was a pilot electrical distribution system functional inspection. This inspection was conducted from February 12 through March 16, 1990. The inspection team reviewed the design and implementation of the plant electrical distribution system and the adequacy of associated engineering and technical support. Overall, the NRC concluded that the design and implementation of the electrical distribution system was adequate, with design attributes well documented, retrievable, and verifiable. Some deficiencies were noted, however, in the design control program. CP&L determined that the identified deficiencies were not indicative of programmatic weaknesses. One deficiency was subsequently determined to be a non-cited violation. The appropriate corrective actions were taken to resolve the identified non-cited violation.

Reactor Auxiliary Building Emergency Exhaust System Issues

On February 7, 1990, while investigating a damper problem in the Reactor Auxiliary Building Emergency Exhaust System (RABEES), personnel questioned why the system was declared inoperable due to isolation damper failures, when doors and hatches which provide a similar isolation function to the ventilated rooms were not controlled, and could be blocked open for maintenance activities. It was concluded that administrative controls should be in place to maintain the pressure boundary for the RABEES. On April 5, 1990, engineering evaluations determined that without further testing the RABEES could not be determined to be capable of performing its design function if doors and hatches were open. As a result, LER 90-010 was issued.

In December 1990, HNP personnel identified that some of the Emergency Core Cooling System (ECCS) piping was located outside of the RABEES boundary. The condition was evaluated and determined not to constitute an unreviewed safety concern because calculations showed that offsite dose would not exceed the 10CFR100 limits. In July 1995, this condition was reevaluated and determined to be a potential unreviewed safety question. The cause of this condition was inadequate documentation and communication of assumptions within the architect engineering firm during initial plant design. The safety significance of this condition was minimal since ECCS piping and components are designed such that large leak rates are not expected and are monitored during normal operation. In addition, normal plant operational, maintenance, and health physics practices would have controlled leakage such that offsite airborne radioactivity following a design basis accident remained within 10CFR100 limits. The approach taken to resolve this issue was to reanalyze dose assessments, with an administrative limit on external leakage for components outside the RABEES boundary. This issue was determined to be an unreviewed safety question and the resolution was submitted to the NRC for approval. This condition was reported in LER 95-006-01 on November 30, 1995. The issue has subsequently been reviewed and approved by NRC.

On January 10, 1996, a pressure boundary door for RABEES was found blocked open. This condition caused the RABEES to be unable to maintain the required differential pressure. This example of inadequate administrative controls was described in LER 96-01 and NOV 96-01-01. As described in the response to the NOV, training was performed as an interim action to re-emphasize the requirements for RABEES boundary doors. Also as described in the response to the NOV, a plant modification will be installed that provides alarming capabilities for RABEES boundary doors.

Generic Letter 89-13 and Service Water System Configuration Control Issues

On July 18, 1994, a single failure vulnerability was identified in the Emergency Service Water System (ESW) in that a single active failure coincident with certain design bases conditions could result in inadequate cooling of the Charging/Safety Injection Pumps (CSIPs). Plant personnel became concerned about the effects of backpressure on ESW flow through the CSIP coolers if one auxiliary reservoir return valve were to fail to open on demand. One ESW return header would reach full pump discharge pressure due to this assumed failure of the auxiliary reservoir return valve . This would then force "hot" water backwards through one Emergency Diesel Generator cooler to both CSIP coolers since the ESW trains were cross connected at the CSIPs. This would result in inadequate heat removal from the CSIPs oil coolers and equipment function could not be guaranteed. This event resulted in a Licensee Event Report (LER 94-003) and an NRC Level III violation (94-21-01). This condition was caused by inadequate identification and

consideration of the effects of an active valve failure on the cross-connections at the CSIP during initial ESW design and operational valve lineup development. This event was determined to have minimal safety significance based, in part, on reasonable expected operator actions. Corrective actions for this event included performance of a Service Water System Operational Performance Inspection (SWOPI) in 1994.

The SWOPI was intended to verify that the ESW system was capable of fulfilling its thermal and hydraulic performance requirements and was operated consistent with its design bases. This inspection used the guidance in NRC Temporary Instruction 2515/118, "Service Water System Operational Performance Inspection." A number of design control issues were identified during the SWOPI. A plan has been developed to complete the corrective action items identified for ESW in the SWOPI. Although some of these corrective action items are not completed, the items are being dispositioned through the Corrective Action Program and the ESR process.

An assessment of Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment, " implementation also identified the need for additional corrective actions for ESW. It was also concluded that the Emergency Service Water would perform its intended safety functions, although further documentation was needed to show its capability at the most limiting environmental conditions.

One issue identified was that the current limiting conditions for operation for the Ultimate Heat Sink would not ensure adequate heat removal in a design basis accident. Interim corrective actions included establishing more conservative limits using administrative controls, (Technical Specification Interpretations), until options for resolution were evaluated. It was determined on May 21, 1996, that these more conservative limits should be established as the limiting conditions for operations and an Operating License amendment request was submitted on October 30, 1996. HNP received an NRC NOV regarding the timeliness of corrective actions for this issue on December 20, 1996.

Another recent configuration control issue involved methods for isolating ESW cooling water supply to an air handler unit in the intake structure. After determining that the cooling water was not required to maintain the required building temperatures, the supply lines were isolated using the equipment clearance program rather than the ESR process. On September 15, 1995, NED informed operations personnel that cooling water isolation to the air handler was being credited in an effort to improve performance margin for ESW and requested that a clearance be placed on the supply valves. This clearance was installed on September 21, 1995. In December 1996, the NRC Resident Inspector identified that this was an inappropriate use of the clearance process (NRC NOV 96-11-06). Resolution of this issue includes an assessment of other outstanding clearances,

caution tags, and Equipment Inoperability Records. Other conditions have been identified involving the potential inappropriate use of the clearance process. These conditions will be resolved through the Corrective Action Program.

The SWOPI also concluded that the Component Cooling Water and Essential Services Cooling Water Systems would achieve their intended safety functions during postulated design bases accident conditions.

Engineered Safety Feature Actuation System/Reactor Protection System

In 1995, the NAS reviewed the "A" Train Engineered Safety Feature Response Time testing. One issue identified inconsistencies among the FSAR, design calculations and PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report." HNP developed an action plan to evaluate FSAR accuracy, completeness, and consistency and to establish necessary programmatic controls for future FSAR fidelity. This FSAR improvement program was initiated in early 1996 and is discussed further in the Improvement Initiatives section of this attachment.

A safety system functional evaluation of the Reactor Trip System was performed in 1996. The evaluation was intended to assess the adequacy of engineering activities, such as modifications, evaluations, and calculations, in maintaining the system's design bases and to evaluate the system design conformance to the applicable sections of the FSAR. This evaluation identified several apparent discrepancies involving completeness and consistency in the Reactor Trip System Design Bases Document. The evaluation also identified several apparent discrepancies in the FSAR description of the Reactor Trip System. Condition Reports were initiated to address these concerns. Each of these Condition Reports has been evaluated for operability and reportability concerns. Although corrective actions are still in progress, the discrepancies do not indicate that the system would not perform its design function.

Missing Swing Pump Breaker Interlocks

In July 1996, during a review of design control and plant changes, the NRC Resident Inspector identified instances in which plant configuration was inconsistent with FSAR and DBD descriptions. This problem involved a failure to install mechanical interlocks to prevent simultaneous start of both a spare and dedicated Component Cooling Water Pump on a single electrical safety bus by an automatic signal. As described in the FSAR, the purpose of the interlocks was to prevent a simultaneous start of both pumps by the Load Sequencer following a loss of offsite power. A simultaneous start of both pumps could result in an overload of the Emergency Diesel Generators. Additionally, there was a failure to install interlocks for Charging/Safety Injection Pump breakers as described in the DBDs. This issue also involved the removal of key interlocks designated for the

spare breaker cubicles referenced in plant drawings without conduct of a 10CFR50.59 evaluation. This event is further described in violations 96-06-01 and 96-06-02 and Licensee Event Report 96-014.

These instances were determined to be of minimal safety significance in consideration of the operational history of HNP. The root cause was determined to be a failure to reconcile plant design with the FSAR. This event reinforced the need to complete the FSAR Improvement Plan described in the Improvement Initiatives section of this attachment and in Attachment G.

E. Improvement Initiatives

1. FSAR Improvement Plan

An FSAR Improvement Plan was developed in May 1996. The plan contains three key elements, an initial review by knowledgeable individuals to resolve questionable information, selected detailed reviews, and implementation of routine use of the FSAR in day-to-day activities.

As of February 4, 1997, approximately 71% of the FSAR sections have been initially reviewed and approximately 275 Condition Reports have been written identifying discrepancies between the FSAR and the actual plant configuration or operation practices. The majority of the discrepancies are considered to be minor technical or editorial problems. Four are considered to be significant in that they involved potential reportability or operability concerns. Three of these four have been discussed in either this attachment or Attachment B, (improper testing of reactor trip bypass breakers, missing interlocks for the Component Cooling Water swing pump and control room ventilation system testing). The fourth Condition Report involves an EDG inoperability concern during testing due to overcurrent relay (51V) design deficiency and was reported in LER 96-023. These Condition Reports are being processed through the Corrective Action Program.

Another example of how this initiative has improved HNP's confidence that equipment configuration and performance meets the design basis is the addition of an FSAR review section in the Post-trip Review. The procedure for post-trip reviews now requires a review to identify discrepancies between the FSAR description and equipment performance during the event.

As discussed in Attachment G, the FSAR improvement plan will be completed.

2. Surveillance Program Review Project

The systematic review conducted for Generic Letter 96-01 and the Surveillance Program Review Project discussed in the Improvement Initiatives Section of

Attachment B will provide additional confidence that SSC configuration and performance are consistent with design bases.

3. Steam Generator Replacement and Power Uprate

HNP is planning to replace steam generators and conduct a power uprate project in the near future. In preparation for this, many of the design calculations will be reviewed and revised. HNP plans to use this opportunity to provide additional confidence in SSC configuration and performance.

F. Conclusion

The rationale for concluding with reasonable assurance that system, structure, and component configuration and performance are consistent with the design bases is based upon the following key elements:

1) Technical Specifications and FSAR assessments performed prior to initial operation;
2) Established programs to maintain the design bases and configuration control;
3) Successful pre-operational and startup testing results;
4) Comprehensive surveillance testing program;
5) Established preventative and predictive maintenance programs;
6) Established post-maintenance and post-modification testing programs; and an
7) Corrective Action Program.

Deficiencies have been periodically identified. However, when their number, type, safety significance, and resulting corrective actions are considered in the whole, these deficiencies do not change our conclusion today. CP&L has addressed and will continue to assess SSC configuration and performance and to address identified deficiencies in accordance with the Corrective Action Program.

IV. H.B. Robinson Nuclear Plant

A. Summary

Existing programs and processes provide confidence that structures, systems, and component (SSC) configuration and performance are consistent with design bases. The RNP self-assessment program in conjunction with the Nuclear Assessment Section and external organizations have been effective in identifying specific problems and programmatic deficiencies as well as establishing specific corrective actions and programmatic improvements.

RNP conducted a Design Basis Reconstitution project between January 1988 and May 1994 as described in Attachment F. The project consisted of structuring the design basis information and supporting calculations/analyses, applicable to selected plant systems, into Design Basis Documents (DBD) and establishing controls for future use. The identification and correction of problems discovered during the DBD Reconstitution project and other improvement initiatives, and the emphasis on maintaining configuration and monitoring performance have improved RNP systems related to their capability of performing their design functions. RNP's ongoing program of self-assessments continues to identify and establish corrective actions for occasional inadequacies discovered in existing configuration and performance. These programs provide added confidence that SSC configuration and performance are consistent with the design bases.

Performance monitoring and testing programs are in place at RNP that periodically evaluate the performance of SSCs related to specified acceptance limits. These monitoring and testing programs provide a base of confidence that SSCs are capable of performing their intended safety functions in accordance with the design bases requirements. These programs are described in the following sections.

In an effort to assess the effectiveness of these programs and processes in connection with the preparation of this response, RNP conducted a review of selected documents prepared between January 1, 1987, and December 31, 1996. Selection was made from the following:

1) Licensee Event Reports,
2) Notices of Violations,
3) Conditions reports,
4) Internal self-assessments, and
5) Audits by internal and external organizations.

The review was designed to identify conditions or trends in the configuration or performance of SSCs related to design bases requirements.

Although instances of inadequate SSC configuration and performance were noted in some documents during the review, no indication of general degradation of individual

structures, systems or components configuration or performance was observed. Corrective actions that addressed specific problems and for broader programmatic issues, as appropriate, were also noted.

RNP initiatives for improvement were also evident during the review. Examples included design bases reconstitution, revision of administrative controls that provide instructions for SSC modification, and specific improvement projects related to both configuration and performance.

Accordingly, RNP concludes with reasonable assurance that, notwithstanding previously identified deficiencies, the configuration and performance of SSCs are generally consistent with design bases requirements. RNP has addressed past discrepancies by correcting the exception and taking action, as appropriate, to preclude repetition. The following sections provide additional detail to substantiate RNP's rationale. In certain areas, RNP has determined that additional reviews or actions are appropriate to provide additional assurance that RNP procedures are reasonably consistent with the design bases. These are identified in Attachment G.

B. Structure, System, and Component Configuration

The Design Bases reconstitution project conducted at RNP is described in Attachment F. RNP procedure revisions, the purpose of which was to improve configuration control, are discussed in Attachment B. In addition, programs and projects have been implemented or are in progress to strengthen control of configuration at RNP. Projects to assist in configuration control and programs to evaluate configuration control are briefly described in the following sections.

1. Equipment Data Base

Beginning in the 1980's RNP initiated a project to develop an Equipment Data Base System (EDBS). The objective was to create a tool to assist in configuration management. The data base, which continues to be expanded, is intended to support procurement, aid in the identification of obsolete parts, support component history files, assist in component/part safety classification, and provide information on related drawings, vendor documents, and procedures.

This project, to date, has developed an equipment listing of tag numbers for plant components; data collection, including equipment name plate data, plant location, and pertinent drawing references; and component parts listing.

The EDBS provides positive contributions to the plants configuration control program.

2. Operational Configuration Control

Requirements are in place to provide for control of plant component configuration during plant operation. As described in Attachment B, plant operating and testing procedures receive review and approval. Operational configuration controls include:

1) Valve line-up sheets,
2) Electrical line-up sheets,
3) Independent verification of component positioning,
4) Locking of selected valves, and
5) Use of equipment clearances.

These controls provide confidence that component configuration will be maintained consistent with design bases requirements.

3. Design Control Process

Although the CP&L design control process is described in Attachment A, an important occurrence at RNP resulted in revisions to improve the process relating to reactor core configuration control. A description of the circumstances and improvements follows.

RNP experienced problems in 1993 during startup testing after refueling, due in part to misconfigured fuel bundles by the fuel fabricator. The reactor was placed in cold shutdown; and, the NRC issued a Confirmatory Action Letter that documented the agreement to complete certain items prior to restart of RNP, including ". . . conduct a detailed root cause analysis of the core loading problems including . . . the misconfiguration of fuel assemblies." The CP&L team formed to investigate the event concluded that the causes for the event were:

> "(1) CP&L failure to ensure that fabricated fuel meets design requirements due to lack of management direction and inadequacies in CP&L's review and evaluation program, and
> (2) Fuel supplier fabrication with incorrect placement of gadolinia rods was caused by inadequacies in procedures, accountability, training, and overchecks."

By letter dated January 28, 1994, CP&L informed NRC that CP&L had completed the items specified in the NRC's Confirmatory Action Letter with the exception of obtaining NRC concurrence to restart. By letter dated February 8, 1994, the NRC stated that the

> "... NRC staff has reviewed the preparations for restart, independently verified CP&L activities via inspection, and affirmed that the CP&L commitments had been complied with and that there were no significant regulatory issues open that would preclude the restart of Unit 2."

As a result of this event, corrective actions to prevent recurrence were established and have been completed. These measures include changes to the performance review program to require that fabricated fuel assemblies meet design requirements, revision of vendor Quality Control procedures that control the overcheck of the fabricated bundles, and revisions to processes and procedures to better handle the fabrication of multiple fuel types.

RNP believes that this event served as lessons learned and has resulted in additional improvements to the design control program. Subsequent to the 1993 problems, RNP altered methods used to control core reloads. The ESR process, described in Attachment A, is now used to manage core reloads and fuel shuffle. The fuel vendor, Siemens Power Corporation (SPC), performs the analyses of record using appropriate design inputs, and confirms that the core conforms to the plant's design basis and applicable regulations. The ESR is closed according to procedure after the core is loaded and verified using RNP procedures. Core performance is verified by performance of a series of procedurally governed physics tests during power escalation.

C. *Structure, System, and Component Performance*

Testing and monitoring programs are used to measure the performance of SSCs. These programs include:

1) Technical specification surveillance testing;
2) Inservice testing and inservice inspection;
3) Performance monitoring in accordance with the NRC maintenance rule (10CFR50.65);
4) Post-maintenance and post-modification testing; and
5) Post-trip reviews.

These testing programs help to confirm that the performance of SSCs remains within acceptable limits. The programs described in the following sections provide a base of confidence that SSCs are capable of performing their intended safety functions in accordance with design bases requirements.

1. Technical Specification Surveillance Testing

The Technical Specifications surveillance testing program provides for periodic testing of selected SSCs such as pumps, valves, and instrumentation to confirm that their actual performance is consistent with the specified acceptance limits. This surveillance testing confirms that the selected SSCs are capable of performing their intended safety functions. For example, ECCS pumps are periodically tested to demonstrate that developed head and flowrate meet design requirements.

The Bases Section of the Technical Specifications contains, or references the source of, specific licensing bases information to support the acceptance limits for surveillance tests. RNP has reviewed the content of the Improved Technical Specification Bases and applicable sections of the UFSAR related to surveillance testing requirements as a part of RNP's conversion to Improved Technical Specifications (ITS). In support of this effort, information incorporated into the ITS Bases Section for surveillance testing requirements included in the ITSs was verified.

The Technical Specifications Review/Upgrade described Attachment B and the efforts described above provide additional confidence that appropriate testing is conducted to monitor SSC performance related to consistency with design bases requirements.

2. Inservice Testing and Inservice Inspection

Inservice testing and inspections performed to comply with ASME Section XI also provide confidence that SSC configuration and performance are consistent with design. This program covers a wide range of safety related components including pumps, valves, piping and supports, the purpose of which is to detect performance degradation. Following program requirements, if testing indicates performance deficiencies, testing frequency is increased, and/or equipment is declared inoperable, and corrective actions are performed.

3. Maintenance Rule

Performance of systems that support safe plant operation is tracked in accordance with the Maintenance Rule per 10CFR50.65. Essential elements of this program include:

1) Establishment and application of risk-significant criteria;
2) Setting structure, system, and component performance criteria;
3) Trending system performance to demonstrate the effectiveness of maintenance activities;
4) Setting system performance goals; and
5) Periodically assessing performance.

The program contains requirements to perform root cause evaluations and implement corrective actions when a system fails to meet established performance criteria. A preventive/predictive maintenance program is in place in order to support reliable equipment performance. RNP expects implementation of the Maintenance Rule requirements to help identify degrading SSC performance prior to exceeding design bases requirements.

4. Post-Maintenance and Post-Modification Testing

Following preventive or corrective maintenance of plant equipment, post-maintenance testing is required to verify that equipment function and performance are properly restored. At the RNP, post-maintenance testing is required to be controlled in accordance with approved procedures which provide the instructions and guidelines for the selection, performance, and documentation of required post-maintenance testing. Acceptance testing is also required after SSCs modifications to compare performance with design bases requirements. Post-modification testing is described in Attachment A.

These programs provide added confidence that structures, systems, and components are within design bases requirements after maintenance activities or changes.

5. Post-Trip Reviews

The Operations Assessment procedure provides instructions for data collection and approval requirements related to a post-trip review of plant equipment performance. This provides additional opportunities to detect equipment performance issues that may be indicative of operation outside the design bases. Deficiencies identified in plant configuration or equipment performance are required to be documented in the Corrective Action Program.

6. Additional Programs

Periodic performance testing is also performed to meet specific reliability or programmatic requirements. Examples include piping inspections for flow accelerated corrosion, fire protection component testing, and Motor Operated Valve (MOV) testing.

D. Assessments and Reviews

In an effort to assess effectiveness of configuration controls and performance, RNP conducted a review of selected documents prepared between January 1, 1987, and December 31, 1996. Selection was made from the following:

1) Licensee Event Reports,
2) Notices of Violation,
3) Condition Reports,
4) Inspection Reports,
5) Reports of audits by internal and external organizations, and
6) Internal self-assessments.

The review was designed to identify conditions or trends in SSC configuration or performance related to design bases requirement implementation. An assessment of the results of this review was then performed to establish a level of confidence that plant configuration and performance are consistent with design bases requirements.

Initially, Licensee Event Reports (LERs) and Notices of Violation (NOVs) associated with the ten highest ranked systems, based on probabilistic risk assessments, were reviewed to identify conditions related to nonconformance between SSC configuration or performance, and design bases. The causes of identified conditions were reviewed to determine trends or common root causes that could indicate loss of programmatic controls designed to prevent such occurrences.

The selected NOVs and LERs were grouped by associated plant system. Some instances of inadequate SSC configuration or performance related to design bases requirements were noted. However, no common equipment related cause was apparent that would lead to a conclusion of general system or component degradation.

The NOVs and LERs associated with the ten highest ranked systems were also grouped by year of occurrence. A review of the events on this basis revealed that a number of important issues involving configuration, such as confirmation of the inadequate Auxiliary Feedwater Pump Net Positive Suction Head (NPSH) and Safety Injection System single failure problems were identified during the DBD Project. In addition, the number of LERs and NOVs increased during periods in which improvement initiatives detected conditions requiring corrective actions.

The documentation review was expanded to include NOVs and LERs prepared between January 1, 1987, and December 31, 1996, without regard to system or probabilistic risk assessment ranking. A total of fourteen Level III and Category A Environmental Qualification (EQ) violations were identified as having been cited during this period. Three Level III violations and one Category A EQ violation were cited in 1987. Seven violations were cited during the period of time when the DBD project and other

initiatives were identifying substantive findings, (i.e., 1988 through 1993). More than half of these seven related in some way to inadequate configuration or performance of structures, systems, or components. Two Level III violations occurred in 1994, none in 1995, and one in 1996. The Level III violations in 1994 were related to design bases requirements translation; however, the Level III violation occurring in 1996 was not. Violations identified in more recent documentation were less significant, i.e., Level IV, and generally involved human performance problems or personnel errors. This was not unexpected given the overall improvement initiatives begun in 1988.

More recently, discrepancies related to configuration control indicated the need for continued improvement. For example, during the course of a routine inspection conducted May 14 through June 17, 1995, as documented in Inspection Report (IR) 95-19, the NRC identified six (collectively Level IV) violations characterized as, "... repeat examples of a chronic equipment configuration control problem ... symptomatic of a potential programmatic issue." Carolina Power & Light agreed and indicated that the "... examples represent a serious management concern relative to equipment configuration control."

The cited violations resulted from a performance deficiency by certain Operations personnel to plan and manage system and component configuration change. The problems were caused by:

1) Incorrectly positioned components due, in part, to failure to follow procedures,
2) Incorrect assumptions,
3) Inattention to detail, and
4) Inadequate procedures.

Corrective actions were developed, entered in the RNP Corrective Action Program, and monitored to completion. These corrective actions included:

1) Immediate cessation of incorrect evolutions,
2) Returning affected components to the correct position,
3) Revision of inadequate procedures involved,
4) Implementation of disciplinary action and counseling of involved personnel, and
5) Changing the personnel makeup of the shift crew involved to strengthen leadership and work management capabilities.

In addition, a Corporate Performance Evaluation Section assessment was conducted during the week of June 19 through 23, 1995, at the request of Operations Unit management. Based on the findings, corrective actions to address the programmatic issue included requiring significantly expanded use of pre-job briefings. The intent of these briefings is to improve communications and coordination, assure each person understands their role in the upcoming evolution, what indicators are to be expected, and what actions to take if indications are not as expected.

There are, however, indications of improvement at RNP. For example, there has been one Level III violation since 1994. In addition, 47 Notices of Violations were cited during the eighteen month period between January 1994 and June 1995; while 19 were cited during the succeeding eighteen month period; i.e., July 1995 through December 1996.

Additional evidence of a trend of improvement appears in other indicators, including the number of LERs and the number of personnel errors. Data indicate that a total of 23 LERs were filed during the period from January 1994 through June 1995. During the subsequent eighteen month period, i.e., July 1995 through December 1996, a total of 14 LERs were filed. Likewise, the data indicate that personnel errors have decreased. For example, during the second quarter, 1994, significant Condition Reports (described in Attachment D) judged to be caused by personnel errors totaled 70; however, no significant condition reports were judged to be personnel error related during the fourth quarter, 1995; ten occurred during the first quarter, 1996; seven in second quarter; ten in third quarter; and seven occurred in the fourth quarter of 1996.

The RNP Nuclear Assessment Section and ongoing self-assessment programs continue to identify and establish corrective action for inadequacies in existing configuration and performance. Increased confidence that RNP systems are capable of performing their design function is provided by the:

1) Identification and correction of problems during the DBD Project and other improvement initiatives,
2) Emphasis on maintaining configuration and monitoring performance, and
3) Apparent decrease in number and significance of problems.

Of particular note were three specific self assessments:

1. Safety System Functional Inspection of the Auxiliary Feedwater System

A self-initiated Safety System Functional Inspection (SSFI) of the RNP Auxiliary Feedwater (AFW) system was completed in 1987. The overall project objective was to assess the ability of the AFW system to perform its safety-related function as intended. The assessment of operational readiness was made by reviewing the functional areas of system design, testing, maintenance, training, and operations. The findings resulting from the SSFI were of three classifications as follows:

1. Areas that represented the most significant concerns and which needed timely attention;
2. Areas in which increased emphasis on improvement and attention to detail could contribute to a higher state of operational readiness; and
3. Areas in which no problems were found, and in some cases which represented strengths and should be recognized.

One significant issue that did result from this inspection was the determination of inadequate AFW pump net positive suction head under certain combinations of pumps and supply. Although initially identified as apparent insufficient NPSH calculations, the SSFI included a concern that the causes of an observed three-pump-operating flow deficiency were not thoroughly understood. The report recommended investigation and verification of actual pump flow capability. Actual investigation and verification of inadequate AFW pump NPSH subsequently occurred during the AFW Design Basis Document effort in 1989. Lack of appropriate attention to and resolution of technical issue questions, compounded by lack of design basis information and other programmatic problems, eventually led to a Level III violation in 1989. However, the identification and correction of these problems provided increased confidence that the AFW system is consistent with design bases requirements.

2. Safety System Functional Inspection of the Reactor Protection System

RNP conducted a self-initiated Safety System Functional Inspection of the Reactor Protection System (RPS) in 1996. The objective of this inspection was to assess the adequacy and operational readiness of the RPS and adequacy of associated engineering and maintenance activities. The assessment included modifications, calculations, and evaluations by comparison with design bases and operational requirements, and the adequacy of testing to determine performance.

The inspection did not identify any concerns that would defeat the ability of the RPS to perform its safety function under design basis conditions. Two strengths, one deficiency, and three weaknesses were identified. The deficiency involved nonconformance to isolation requirements regarding the protection/control interface of overtemperature/overpower delta T protection channels. A Condition Report (CR) was prepared requiring preparation of an Engineering Service Request (ESR) for additional evaluation. The ESR has been written; and, the evaluation is continuing.

3. Control Room Habitability Assessment

In 1994, CP&L's Nuclear Assessment Department conducted an assessment of the operational performance of Control Room Habitability at RNP. The effort was accomplished through a technical review of compliance commitments, regulations, change documents, and other licensing and design basis information, as well as plant procedures, drawings, and other descriptive documents. In addition, interviews were conducted and performance based, real time observations were performed.

Issues involving incomplete, incorrect, and inconsistent engineering and design documentation, and deficiencies in the implementation of licensing commitments were found. These included identification of insufficient updating of some documents (including the UFSAR), training, and programs affected by plant modifications.

Improvements made as a result of this self-initiated assessment provide increased confidence that the control room will be habitable during adverse conditions and that procedures and design documentation accurately reflect conditions at RNP.

E. Improvement Initiatives

1. Updated Final Safety Analysis Report Review Program

An effort to perform a qualitative review of the information in the UFSAR began in 1996. The program objective is to reestablish a reasonable level of assurance that the information properly reflects RNPs design, construction, and operation.

Although not designed to verify each statement of fact in the UFSAR, the program does require a review of the information contained in text and tables and comparison with current knowledge of specific design configuration and operation of structures, systems, and components.

The Program provides review criteria and requires documentation of discrepancies other than minor editorial or typographical in accordance with the Corrective Action Program described in response to NRC item (d).

2. Self-Assessment

Assessment programs have been developed and implemented at RNP to identify discrepancies and trends that may provide early indication of potential problems in maintenance of the design bases and configuration.

The Nuclear Assessment Section is responsible for:

1) Independent reviews of changes to the facility and procedures as described in the UFSAR, and tests not described in the UFSAR; and
2) Formal assessments of facility operations, corrective actions, Quality Assurance Program activities, and other activities as described in Section 6 of the RNP Technical Specifications.

In addition, RNP established an ongoing program of self-assessments in 1994. The objective is to achieve a higher standard of quality and performance by identification of deficiencies (or areas for enhancement) and implementation of improvements in process, programs, and activities. Each organization unit is required to develop an annual self-assessment plan that must include an evaluation of areas such as the unit's corrective action program and implementation of the procedure development/change process.

These assessments, associated findings, and implemented corrective actions have provided added confidence in the adequacy of the target systems or programs, as well as providing a level of confidence in the effectiveness of the governing procedures and processes. These assessments have assisted, and continue to assist, in identification and resolution of discrepancies.

F. Conclusion

The rationale for concluding that structure, system, and component configuration and performance are consistent with the design bases is based on the following key elements:

1) Programs are established to maintain the design bases and configuration control;
2) The implemented surveillance testing programs objective is to identify potentially inadequate performance;
3) Preventive and predictive maintenance programs are established and implemented;
4) Post maintenance and post modification testing programs are established and implemented; and,
5) A corrective action program, designed to document discrepancies and track corrective actions to completion, is functioning.

Although deficiencies are occasionally identified, the configuration and performance of SSCs are generally consistent with design bases requirements. CP&L has addressed and will continue to address exceptions to this conclusion by correcting the exceptions and taking action, as appropriate, to prevent recurrence.

Attachment D

Processes for Problem Identification and Corrective Action Implementation

I. INTRODUCTION 1

II. CORRECTIVE ACTION PROGRAM 1

A. PROCESS FOR IDENTIFICATION OF PROBLEMS 1
B. ACTIONS TO DETERMINE EXTENT OF PROBLEM 2
C. ACTIONS TO PREVENT RECURRENCE 3
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III. OTHER PROGRAMS 3

A. SELF-ASSESSMENT PROGRAM 3
B. INDEPENDENT ASSESSMENT PROGRAM 4
C. CORRECTIVE MAINTENANCE PROGRAM 4
D. PROCEDURE CHANGE PROGRAM 5
E. DRAWING CHANGE PROGRAM 5
F. OPERATING EXPERIENCE PROGRAM 6
G. EMPLOYEE CONCERN PROGRAM 6

I. Introduction

This attachment provides information in response to Action Item (d):

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

Each of CP&L's three nuclear plants has programs in place for the identification and correction of problems, including actions to determine the extent of problems, actions to prevent recurrence, and appropriate reporting to the NRC. These programs consist of the Corrective Action Program, augmented by several additional programs focused on specific types of problems. These programs include the:

1) Self-Assessment Program;
2) Independent Assessment Program;
3) Corrective Maintenance Program;
4) Procedure Change Program;
5) Drawing Change Program;
6) Operating Experience Program; and
7) Employee Concern Program.

With the exception of the Independent Assessment Program, each of these programs is governed by individual plant procedures and is implemented by line management at each of the nuclear plants. The Independent Assessment Program is governed by corporate procedures.

These processes and programs are changed from time to time to strengthen or provide adjustments, as required, to reflect improvements or organizational changes.

II. Corrective Action Program

A. Process for Identification of Problems

The Corrective Action Program describes the primary process used to document, investigate, and correct conditions adverse to quality. For example, identified discrepancies between plant configuration or operation and the Updated/Final Safety Analysis Reports (U/FSARs) should be documented, evaluated, and resolved in the Corrective Action Program. Each employee working in support of CP&L's nuclear facilities is responsible for noting and reporting conditions adverse to quality. The following discussion summarizes the procedural requirements of the program.

Any CP&L employee or contractor may document an observed condition adverse to quality through initiation of a Condition Report (CR). Each plant has implemented an electronic CR initiation process. A manual method of CR initiation is also provided. Following initiation, supervisory review is required of the CR for potential operability and/or reportability concerns.

CAP personnel are responsible for performing a quarterly review and analysis of adverse conditions, from a site-wide perspective, for potentially adverse trends. CAP subprogram coordinators should assist CAP personnel in evaluating apparent adverse trends affecting their respective subprograms. If data analysis indicates an adverse trend, site procedures direct the initiation of a CR to document the trend so that it may be evaluated and appropriately resolved.

B. *Actions to Determine Extent of Problem*

CRs are classified based, in part, on the potential consequences, significance, and potential recurring nature of the problem. Specific criteria for determining CR level are contained in individual plant procedures.

The Robinson Nuclear Plant (RNP) and Harris Nuclear Plant (HNP) use a three-level program involving Significant, Minor (HNP)/Non-Significant (RNP), and Improvement CRs. The Brunswick Nuclear Plant (BNP) also uses a three-level program, but classifies its CRs into Significant, Important, or Minor CRs. At BNP, conditions which do not meet the criteria for significant or important conditions, but still require corrective action and trending are classified as Minor CRs. BNP's Significant and Important CRs are analogous to Significant CRs at the other two sites. Minor/Non-Significant CRs are evaluated to determine the nature of the condition and appropriate corrective actions, as well as to gather sufficient information to allow problem trending. Improvement CRs are used to identify opportunities for performance improvements and are not intended to document conditions adverse to quality.

Any individual identifying a condition that meets the definition of an adverse or improvement condition (or minor condition at BNP), is responsible for initiating a CR and providing it to a supervisor for review. Site procedures specify the supervisor's responsibilities to review the CR, initiate any immediate corrective actions which are appropriate, determine if the condition represents a potential operability or reportability concern, and approve the CR. If the condition represents a potential operability or reportability concern, Operations should be promptly notified so that they can perform operability and reportability reviews and take appropriate actions in accordance with site procedures. (Operations notification should be made at any point that an individual identifies a potential operability or reportability concern.) Licensing/CAP personnel are responsible for confirming the reportability assessments, as prescribed in site procedures. If the condition does not represent a potential reportability concern, but is a serious or significant event, it may be reviewed with plant management and other appropriate site

personnel as specified in site procedures to determine the significance, and may result in remedial or interim actions until assigned personnel can investigate the identified concern. Standard, controlled processes are defined in site procedures for other less significant conditions identified through the Corrective Action Program.

C. *Actions to Prevent Recurrence*

Significant and Important CRs are evaluated to identify the root cause(s) of identified conditions except where no value is added by additional review. These evaluations are approved by management as required by plant procedures. These evaluations include previous internal and industry operating experience. In addition, extent of condition, root cause, immediate corrective actions and corrective actions to prevent recurrence are required to be identified. Evaluations and corrective actions for CRs are tracked using each plant's Corrective Action Program database. Changes and extensions to evaluations and corrective action completion are controlled by plant procedures and require an appropriate level of management approval.

CR data are periodically analyzed to identify trends, recurring conditions, and areas for improvement. This analysis may be performed by the plant corrective action program staff, or by the individual plant organizational units. A variety of analysis methods (e.g., common cause analysis and symptom classification) are typically used to identify potential trends. Trends and identified areas for improvement are addressed through a variety of methods, such as initiation of a separate CR to evaluate the trend or performance of a self-assessment.

D. *NRC Reportability*

CRs identifying potential operability or reportability concerns are required to be forwarded to the Control Room for operability reviews, consideration of necessary compensatory measures (e.g., Technical Specification action statements) and/or immediate notifications to the NRC (i.e., 10CFR50.72 1-hour or 4-hour reports). At BNP, CRs are also reviewed by the Licensing/Regulatory Programs Unit to determine if any other reporting requirements, such as TS Special Reports, 10CFR21 evaluations, or 10CFR50.73 Licensee Event Reports (LERs), are required. CRs that are determined upon initiation or initial review to identify potentially reportable conditions are initially classified as Significant at RNP. At HNP, the Corrective Action Program and licensing staffs evaluate Significant and Minor CRs for reportability.

III. Other Programs

A. *Self-Assessment Program*

1) Another process for problem identification at the three nuclear plant sites is the self-assessment program. The self-assessment program at each site requires individual

line organizations to develop annual self-assessment plans and approve completed self-assessments. Plant procedures recommend that one self-assessment be performed per calendar quarter. Self-assessment topics are determined based upon criteria such as identified weaknesses, impact on nuclear safety, and program or process changes.

Details of the assessment process, including the requirements for planning, preparation, conduct, and reporting of results to management, are contained in plant procedures.

B. *Independent Assessment Program*

The Independent Assessment program is implemented at each site primarily by the Nuclear Assessment Sections (NAS). Each of the NAS organizations performs several assessments per year. These assessments include those required by regulatory commitments, as well as discretionary assessments. Corporate procedures include requirements for assessment planning, reporting results to management, and follow-up. Written responses to NAS-identified issues are required and are addressed to the senior Nuclear Generation Group management. Issues identified in NAS assessments are also included in the plant Corrective Action Program for problem evaluation, correction, and trending.

An additional component of the overall CP&L Self-Assessment and Independent Assessment Programs is the corporate Performance Evaluation Support Unit (PES). The PES unit performs independent assessments of the line organization's self-assessment and Corrective Action Programs and assessment of the plant NAS organizations. The PES unit also leads self-assessments in each major functional area (e.g., maintenance, operations, and engineering) every 24 months. Written responses are prepared for regulatory required PES assessments, such as the assessment of the site NAS organizations. Other PES assessment requirements are included in corporate procedures.

C. *Corrective Maintenance Program*

The corrective maintenance program provides a mechanism to document and correct plant equipment deficiencies. When deficiencies are identified, corrective maintenance is performed to restore the equipment to specifications. Appropriate post-maintenance testing determines the effectiveness of performed maintenance. Plant personnel are responsible for identifying plant equipment deficiencies through initiation of Action Requests (ARs) at RNP, Trouble Tags (TTs) at BNP, or Deficiency Log Entries (DLEs) at HNP. Review of these items for safety and operability concerns is the responsibility of Operations personnel in the Work Control Center. Work Control Center personnel also are responsible for assigning initial priorities to these items. Planning and scheduling is dependent upon the significance or impact of the necessary work. Required reviews determine if the identified work may be completed as Minor Maintenance,[1] or Tool

[1] Minor maintenance is defined as plant maintenance that has minimal or no impact on system operation or availability as defined by criteria contained in applicable plant procedures and may require documentation.

Pouch Work,[2] or if they need to be escalated to Work Requests (WRs) at HNP and RNP, or Work Request/ Job Orders (WR/JOs) at BNP.

Procedures define site requirements for maintenance tasks and associated work package preparation. These requirements include, as applicable, appropriate reviews, permits and requests, work instructions to a level of detail appropriate to the complexity of the task, a description of support requirements, identification of appropriate materials and special tools, a listing of Post Maintenance Testing Requirements (PMTR) and Radiation Work Permit (RWP) requirements. Site procedures direct the implementor to perform the work and complete the appropriate PMTR and required work documentation.

Significant equipment deficiencies should be documented and evaluated under the Corrective Action Program. This includes, for example, functional failures of systems within the scope of 10CFR50.65 (i.e., the "Maintenance Rule"). Evaluation, correction, and NRC reporting of these conditions is as described under the Corrective Action Program and Reportability Determination Procedures.

D. *Procedure Change Program*

Controlled processes are in place to accomplish plant procedure changes. Sources that identify required procedure changes include (a) procedure usage, (b) CR evaluations, or (c) changes identified while revising other site documents. A summary description of the procedure change process at each site is provided in Attachment B. Where a procedure deficiency results in another adverse condition (e.g., an error in a procedure results in incorrect operation or maintenance), the procedure problem should be documented on a CR. Actions are specified in procedures to determine the extent of the problem, to correct the problem, and to determine reportability to the NRC. Document Services personnel have the responsibility for tracking procedure revisions.

E. *Drawing Change Program*

The process by which required changes are made to plant drawings is defined in the drawing change program. The details of this process are described in site and Nuclear Generation Group procedures. Significant discrepancies between plant configuration and plant design may be documented on Engineering Service Requests (ESRs) or may be addressed through the Corrective Action Program (CAP). This may occur when the discrepancy is identified or during engineering review of the drawing change request. The ESR procedure provides the responsible engineer with screening questions to help identify impacts to plant programs and organizations and required reviews. This information may be utilized in developing the need for ESR team formation and membership, obtaining additional design inputs (DI), and other necessary reviews of

[2] Tool Pouch Work involves simple tasks that have no impact on system operation or availability. Accordingly, these tasks may be performed with or without AR/TT/DLE documentation. Tool Pouch work is defined by site procedures.

proposed plant changes. The design change process is described in Attachment A. Evaluation, correction, and NRC reporting of these conditions are as described under the ESR and Corrective Action Programs.

F. *Operating Experience Program*

CP&L has established Operating Experience (OE) programs by site-specific procedures at its nuclear sites to provide the methodology for review and evaluation of relevant industry operating experiences. The primary objective of the program is to review and process operating experience information and provide a mechanism for issuing information to potentially affected plant units.

The program provides for source document receipt, processing (screening, evaluation, action tracking), and record maintenance of OE item disposition. It designates responsible personnel to help assure that operational type information originating both from within and outside the company is screened, disseminated, and actions are tracked. It also identifies personnel responsible for helping ensuring that those items screened for evaluation are forwarded to cognizant plant personnel.

The Operating Experience Program provides the process for assessing operating experiences from industry sources for possible impact on the operation of CP&L nuclear plants, and it provides the mechanism for the sharing of OE information among CP&L's nuclear sites. Where action is required, corrective actions are initiated to eliminate or reduce the probability of similar incidents. The program also disseminates appropriate information of importance to affected groups.

The OE Program includes, but is not limited to:

1) Applicable INPO operating experience reports and documents;
2) NRC Information Notices and other applicable documents;
3) Significant Adverse Condition Reports generated within the company.

G. *Employee Concern Program*

The employee concern program, known as Quality Check, provides a means for CP&L and contractor employees to report observed or perceived concerns to management in a confidential manner. Concerns that are received are to be evaluated with appropriate corrective actions.

Attachment E

Assessment of Overall Effectiveness of Current Processes and Programs

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II. REVIEW OF REQUESTED DESIGN BASES INFORMATION 1

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B. TRANSLATION OF DESIGN BASES REQUIREMENTS INTO PROCEDURES 2
C. CONFIGURATION AND PERFORMANCE CONSISTENCY WITH DESIGN BASES 2
D. PROCESSES FOR PROBLEM IDENTIFICATION AND CORRECTIVE ACTION 3
E. DESIGN BASIS DOCUMENT PROGRAM 4

III. USE OF SELECTED PERFORMANCE INDICATORS 4

A. CP&L 5
B. INSTITUTE OF NUCLEAR POWER OPERATIONS / WORLD ASSOCIATION OF NUCLEAR OPERATORS 5
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I. Introduction

This attachment provides information in response to Action Item (e):

(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.

To determine the overall effectiveness of the current processes and programs in concluding that the configurations of the plants are consistent with the design bases, CP&L:

1) Reviewed the information presented in Attachments A through D and F;
2) Considered use of selected performance indicators;
3) Reviewed provisions for management assessments.

II. Review of Requested Design Bases Information

A. Engineering Design and Configuration Control Processes

Attachment A described the engineering design and configuration control processes, including those that implement 10CFR50.59, 10CFR50.71(e), and Appendix B to 10CFR50. In particular, Attachment A:

1) Identified the key design documents that exist to support plant operation (Section II);
2) Summarized how Engineering uses configuration management to control the physical condition of the plant with respect to the design bases (Section III.B);
3) Acknowledged the use of approved procedures to administratively control the design change process (Section III.C);
4) Provided detailed descriptions of various engineering information that is used in the design change process (Section III.D);
5) Discussed the 10CFR50.59 process as it relates to engineering design and configuration control, including the screening step and the Unreviewed Safety Question Determination (Section IV);
6) Addressed the procedural requirement for updating the U/FSAR for design changes in accordance with 10CFR50.71(e) (Section V);
7) Discussed aspects of the Quality Assurance Program relative to engineering design and configuration control (which provides the administrative basis for conformance to Appendix B to 10CFRPart50) (Section VI).

These areas are subject to assessments. When conditions adverse to quality are identified, they are required to be addressed via the Corrective Action Program. CP&L considers the process of assessing performance and correcting problems to be

fundamental to the proper implementation of engineering design and configuration control processes.

Upon review of the programs and processes, as outlined in Attachment A, CP&L concludes that the programs contain the features necessary for an effective engineering design and configuration control process.

B. *Translation of Design Bases Requirements Into Procedures*

Attachment B described the rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures. The specific rationale for each plant was unique, but they shared a common reasoning. As described in Attachment B, each plant:

1) Administratively controls the translation of design bases requirements into procedures (Sections II.B, III.B and IV.B);
2) Has subjected itself or been subjected to a number of assessments and reviews from both internal and external sources (Sections II.C, III.C and IV.C);
3) Has undertaken several procedure improvement initiatives, of which some are yet to be completed. See Attachment G. (Sections II.D, III.D and IV.D).

Recently completed, ongoing, and planned improvement initiatives are considered evidence that CP&L is committed to continued improvement in these areas which are subject to further assessments. When conditions adverse to quality are identified, they are required to be addressed via the Corrective Action Program. CP&L considers the process of assessing performance and correcting problems to be fundamental to the proper translation of design bases requirements into operating, maintenance, and testing procedures.

C. *Configuration and Performance Consistency With Design Bases*

Attachment C described the rationale for concluding that the system, structure, and component configuration and performance are consistent with the design bases. The specific rationale for each plant was unique, but they shared a common reasoning. As described in Attachment C, each plant:

1) Administratively controls the configuration of systems, structures, and components (Sections II.B, III.B and IV.B);
2) Tests and monitors the performance of systems, structures, and components (Sections II.C, III.C and IV.C);
3) Has subjected itself or been subjected to a number of assessments and reviews from both internal and external sources (Sections II.D, III.D and IV.D);
4) Has undertaken several improvement initiatives, of which some are yet to be completed. See Attachment G (Sections II.E, III.E and IV.E).

Recently completed, ongoing, and planned improvement initiatives are considered evidence that CP&L is committed to continued improvement in these areas which are subject to further assessments. When conditions adverse to quality are identified, they are required to be addressed via the Corrective Action Program. CP&L considers the process of assessing performance and correcting problems to be fundamental to maintaining the configuration and performance of systems, structures, and components consistent with design bases.

D. Processes for Problem Identification and Corrective Action

Attachment D described the processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence, and reporting to NRC. In particular, Attachment D described the:

1) Process for the identification of problems (Section II.A);
2) Actions to determine the extent of the problem (Section II.B);
3) Actions to prevent recurrence (Section II.C);
4) Reporting to the NRC (Section II.D);
5) Self-Assessment Program as an additional process for the plant to identify problems (Section III.A);
6) Independent Assessment Program as an additional process for NAS to identify problems (Section III.B);
7) Corrective Maintenance Program as the mechanism to document and correct plant equipment deficiencies (Section III.C);
8) Procedure Change Program as the mechanism to change plant procedures (Section III.D);
9) Drawing Change Program as the mechanism to change plant drawings (Section III.E);
10) Operating Experience Program to provide review and evaluation of relevant industry operating experience (Section III.F);
11) Employee Concern Program (i.e., Quality Check) as a means to report observed or perceived concerns to management in a confidential manner (Section III.G).

Upon review of the administrative control programs, as outlined in Attachment D, CP&L concludes that the programs contain the features necessary to identify problems and implement corrective actions (including actions to determine the extent of problems, actions to prevent recurrence, and reporting to NRC).

E. Design Basis Document Program

Attachment F described the Design Basis Document Program. The specific description for each plant was unique, but they shared a common presentation. In particular, Attachment F:

1) Described the design bases information review programs (Sections II.A&E, III.A&E, and IV.A&E);
2) Identified the systems, structures, and components (SSC) included in the DBDs (Sections II.B, III.B and IV.B);
3) Identified plant-level topics (e.g., seismic qualification, consideration of high-energy and moderate-energy line break) included in the DBDs (Sections II.B and III.B);
4) Described the process to provide correctness and availability of the design bases information (Sections II.C, III.C and IV.C); and
5) Described how the DBDs are maintained current (Sections II.D, III.D and IV.D).

These areas are subject to assessments. When conditions adverse to quality are identified, they are to be addressed via the Corrective Action Program. CP&L considers the process of assessing performance and correcting problems to be fundamental to the proper implementation of design bases.

Upon review of the administrative control programs, as outlined in Attachment F, CP&L concludes that the programs contain the features necessary for an effective design basis document control process.

III. Use of Selected Performance Indicators

The responsibilities of CP&L management include monitoring the results of and trends in both internal and external (e.g., NRC and INPO/WANO) performance indicators.

Performance should be viewed collectively, rather than focusing upon a particular indicator exclusively, to obtain an understanding of overall performance. The intent is to use the indicators as a tool for identifying areas that need additional management attention.

The selected performance indicators were considered to be more directly applicable to engineering, operations, and maintenance with respect to design bases, configuration control, and equipment performance. The results and trends of the following performance indicators have been considered in assessing the overall effectiveness of CP&L's current processes and programs in concluding that the configuration of the plants are consistent with the design bases:

A. CP&L

1) Unit Capability Factor / Unit Capacity Factor
2) Unplanned Capability Loss Factor
3) Unplanned Automatic Scram
4) Safety System Actuations
5) High Pressure Coolant Injection System Unavailability
6) Residual Heat Removal System Unavailability
7) Emergency AC Power Unavailability

B. Institute of Nuclear Power Operations / World Association of Nuclear Operators

1) Unit Capability Factor
2) Unplanned Capability Loss Factor
3) Unplanned Automatic Scrams
4) Safety System Performance

In November, HNP and RNP joined BNP on INPO's list of plants that have been categorized as Excellent. The excellence list recognizes plants with outstanding performance in operational safety and reliability. CP&L is the only utility in the nation with more than two nuclear plants on the current excellence list.

C. NRC

1) Automatic Scrams While Critical
2) Safety System Actuations
3) Significant Events
4) Safety System Failures
5) Forced Outage Rate

IV. Management Assessments

A. Plant Nuclear Safety Committee (PNSC)

The purpose of the PNSC for each plant is similar. In particular, the PNSC:

1) Is established as an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety (BNP and RNP);
2) Functions to advise the Plant General Manager on all matters related to nuclear safety (HNP).

The composition of the PNSC for each plant includes members who represent:

1) Operations;
2) Maintenance;
3) Engineering;
4) Health Physics/Chemistry;
5) Regulatory Affairs;
6) Nuclear Assessment.

Some of the specific activities of the PNSC include, but are not limited to:

1) Review of all reportable events;
2) Review of plant operations to detect potential nuclear safety hazards;
3) Performance of special reviews and investigations, and reports thereon as requested by the Manager - Nuclear Assessment Section;
4) Review of all proposed changes to the Technical Specifications;
5) Review of all proposed procedures, tests, or experiments that constitute an unreviewed safety question, or involve a change to Technical Specifications;
6) Review of all proposed modifications that constitute an unreviewed safety question or involve a change to the Technical Specifications.

B. Nuclear Safety Review Committee (NSRC)

The NSRC adds valuable perspective to the oversight function. The NSRC is chartered:

1) To advise the Vice President of the respective plant on the adequacy and implementation of the nuclear safety policies and problem solutions;
2) To provide a strong outside nuclear industry perspective to the plant's nuclear safety performance; and
3) To provide an independent source of nuclear safety information for management.

The composition of each plant NSRC includes, but is not limited to:

1) Senior management, such as:
   a) Vice President of the respective plant;
   b) Vice President - Nuclear Engineering;
   c) Manager - Nuclear Assessment; and
   d) Plant General Manager;
2) Two outside nuclear experts.

Some of the specific overview responsibilities of the NSRC include review of:

1) All reportable events;
2) Safety evaluations of significant changes in the facility as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report, and test or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10CFR50.59(a)(1);
3) Proposed changes to procedures, equipment, systems or the facility, and selected proposed tests or experiments which involve an unreviewed safety question as defined in 10CFR50.59(a)(2);
4) Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;
5) Significant, recognized indications of an unanticipated deficiency in some aspect of design or operation of systems, structures, or components that could affect nuclear safety;
6) Significant inspection and evaluation reports (e.g., SALP), including CP&L's responses, performed by regulatory authorities including NRC, INPO, ANI, and state regulatory bodies;
7) Significant assessment reports and issues from Nuclear Assessment.

V. Conclusion

CP&L has reasonable assurance that the current processes and programs for maintaining consistency between the configuration of CP&L plant(s) and the design bases are adequately effective overall. In reaching this conclusion, CP&L considered information such as that presented in Attachments A through D and F, about:

1) Applicable processes, programs and procedures;
2) The results of various applicable assessments and reviews;
3) Various applicable improvement initiatives; and
4) The design basis document effort.

Some additional improvement initiatives are planned to strengthen our processes and programs, as summarized in Attachment G.

Deficiencies identified in the future through assessment programs or otherwise are expected to be corrected by the Corrective Action Program at each plant.

Attachment F

Design Bases Reconstitution Type Activities

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I. Introduction

This attachment provides information in response to the following request:

In responding to items (a) through (e), indicate whether you have undertaken any design review or reconstitution programs, and if not, a rationale for not implementing such a program. If design review 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.

This attachment responds to this request by describing the programs for the compilation of design bases information at the three CP&L nuclear plant sites that have resulted in a set of Design Basis Documents (DBDs) for each plant. Included as separate sections for each of the three plants are:

1) A description of the design bases information review programs;
2) Identification of the systems, structures, and components (SSC) included in the DBDs;
3) Identification of plant-level topics (e.g., seismic qualification, consideration of high-energy and moderate-energy line break) included in the DBDs;
4) A description of the process to provide correctness and accessibility of the design bases information; and
5) A description of how the DBDs are maintained current.

Note: Other attachments in the response were arranged alphabetically (i.e., BNP, HNP, RNP), but due to the chronological evolution of the DBD program, this attachment was arranged chronologically (i.e., RNP, BNP, HNP).

In addition to the DBD program described below, CP&L presented a plan to review the information in the Updated/Final Safety Analysis Reports (U/FSARs) for the plants to representatives of the NRC during a meeting held in the NRC headquarters offices on May 30, 1996. Also, the completion of prior design bases-related commitments and future enhancements (see Attachment G) will provide further assurance of the correctness and accessibility of design bases information in the U/FSAR and other key documents.

II. Robinson Nuclear Plant Design Basis Documents

A. *General Description*

The Robinson Nuclear Plant (RNP) "Design Basis Reconstitution Project" was conducted between January 1988 and May 1994. A description of the plan for this effort was submitted to the NRC by letter dated May 3, 1988, "Plan and Schedule for Design Basis Reconstitution." The purpose of this effort was to structure the design bases information and supporting calculations/analyses, applicable to selected plant systems, into Design Basis Documents (DBDs), and to control them for future use. The general steps identified in the plan for preparing the DBDs for selected plant systems included:

1) Acquire the original system design bases information;
2) Integrate regulatory commitments made after the issuance of the operating license (OL);
3) Document the current system design bases;
4) Document and/or acquire the original system calculations/analyses;
5) Identify the current system calculations/analyses;
6) Identify and catalogue current system descriptive information;
7) Integrate the information in steps 3, 5, and 6 above into a draft system DBD;
8) Perform interdiscipline design verification of the information in the draft DBD;
9) Approve the design verified system DBD and issue it as a controlled document on a preliminary use basis;
10) Transmit copies of the documents referenced in an approved system DBD to plant document control;
11) Perform a procedure and hardware inspection of the subject system using information in the approved system DBD and its references as the inspection's standard; and
12) Revise the system DBD to incorporate improvements identified by the procedure and hardware inspection and reissue it for use.

The project comprised two phases. Phase 1 was the selection of the plant safety-related systems and topics to be included in the DBDs and the compilation of design bases information for the selected systems and topics into DBDs. Phase 2 involved the validation of system safety-related design bases information and resolution of discrepancies. Although the term "reconstitution" was used to describe the program, the RNP program was primarily a compilation of existing design bases information. However, two significant reconstitution efforts, related to electrical distribution and instrument setpoints, were initiated and completed for RNP. These efforts primarily involved performing new calculations.

Since the RNP effort was initiated prior to the establishment of industry guidelines (i.e., NUMARC 90-12, "Design Basis Program Guidelines"), a pilot program approach was

utilized to help set the RNP standard. The project was focused on acquiring design bases and descriptive information for selected safety-related systems and topics and integrating that information into a set of DBDs.

The safety-related system DBDs were developed, reviewed, and approved by CP&L, with input from the RNP Nuclear Steam Supply System (NSSS) vendor (Westinghouse) and the RNP Architect/Engineer (Ebasco). The original system design bases at the time of operating license issuance was established first, and then any post-OL design bases changes, usually resulting from regulatory commitments, were added to establish the current design bases. A total of 18 system DBDs and 11 plant-level topical or Generic Issues Documents (GIDs) have been created for RNP.

B. Program Scope

The 18 system DBDs are as follows:

DBD-SD-01	Reactor Coolant
DBD-SD-02	Safety Injection
DBD-SD-03	Residual Heat Removal
DBD-SD-04	Service Water
DBD-SD-05	Emergency Diesel Generator
DBD-SD-06	Reactor Safeguards/Protection
DBD-SD-10	Incore/Excore Instrumentation
DBD-SD-13	Component Cooling Water
DBD-SD-16	Electrical Power Distribution
DBD-SD-19	Radiation Monitoring
DBD-SD-21	Chemical & Volume Control
DBD-SD-25	Main Steam
DBD-SD-26	Condensate System
DBD-SD-27	Feedwater
DBD-SD-32	Auxiliary Feedwater
DBD-SD-36	Post-Accident HVAC
DBD-SD-51	Reactor Vessel Level Instrumentation
DBD-SD-62	Cable and Raceway

The 11 Generic Issues Documents (GIDs) are as follows:

DBD-GID-01	Environmental Qualification
DBD-GID-02	10CFR50 Appendix R (Fire Protection)
DBD-GID-03	Seismic Qualification
DBD-GID-04	ALARA and Radiation Shielding
DBD-GID-05	Masonry Fill
DBD-GID-06	Pipe Failures
DBD-GID-07	Hazards Analysis
DBD-GID-08	Regulatory Guide 1.97 (Instrumentation)
DBD-GID-09	Emergency Response Facility Information System/Safety Parameter Display System
DBD-GID-13	Single Failure
DBD-GID-RCI	Reactor Containment Isolation

C. DBD Correctness and Accessibility

The RNP DBDs are based on referenced source documents that are listed throughout the DBDs and that are generally maintained as records within the Quality Assurance Program records control process. The DBDs/GIDs are controlled documents that are available in the plant and corporate libraries.

D. *Maintaining DBDs Current*

The RNP DBDs are considered "design documents," and as such they are required, by CP&L procedures, to be maintained in a controlled format that is useable by CP&L organizations. Changes to the DBDs are tracked in the Nuclear Revision Control System (NRCS), which is CP&L's configuration management tool for design documents. NRCS allows for maintenance of the current status of design documents and control of outstanding changes in a consistent and timely fashion. Nuclear Engineering Department (NED) procedures require that the DBDs be considered during preparation of plant modifications/design document changes, and that they be revised if the plant modification/design document change impacts the design bases information in the affected DBDs. Also, CP&L's Engineering Support Personnel (ESP) Training Program provides training on DBD use, and the Corrective Action Program (CAP) provides the mechanism for documenting, investigating, and correcting conditions adverse to quality, including identified discrepancies between plant configuration or operation and the Updated/Final Safety Analysis Report (U/FSAR).

E. *Program History*

The 18 system DBDs were initially drafted from 1988 through 1993, and the 11 GIDs were initially drafted from 1990 through 1994. The safety-related system DBDs included the system functional requirements, the regulatory requirements that affect system design, and the codes/standards applicable to the system. The types of information typically included are the "what" and "why" of a system's design bases (i.e., "what" refers to the actual design bases information and "why" refers to the supporting design information that provides the rationale for the design bases[1]):

So that the safety-related system DBD requirements would be substantially correct, a DBD validation[2] was performed based on guidance in CP&L procedure DBD.PII.2, "Design Basis Document Validation," Revision 0 and Revision 1. Validation was performed on the system's critical design parameters as they relate to the system hardware, performance, and configuration. The validation process for safety-related system DBD requirements included some or all of the following attributes: system configuration (including walkdown[3], if necessary), performance, instrumentation, setpoints, surveillance tests, operating and emergency procedures, maintenance procedures, and vendor technical manuals relevant to the system being validated. Upon completion of the validation of a Revision 0 safety-related system DBD, the validation feedback was incorporated into Revision 1 of the DBD. Note that not all of the DBDs/GIDs were validated. Specifically, 14 of the 18 system DBDs were validated.

1 NUMARC 90-12, "Design Basis Program Guidelines" dated October 1990.

2 Validation - A systematic approach to assure that the draft design bases information is substantially correct (DBD.PII.2).

3 Walkdowns were performed only to the degree necessary to review existing site conditions to substantiate a statement in the associated DBD as determined necessary by the validation team.

The 14 validated safety-related system DBDs were Reactor Coolant, Safety Injection, Residual Heat Removal, Service Water, Emergency Diesel Generator, Reactor Safeguards/Protection, Incore/Excore Instrumentation, Component Cooling Water, Electrical Power Distribution, Chemical and Volume Control, Auxiliary Feedwater, Post-Accident HVAC, Reactor Vessel Level Instrumentation, and Cable and Raceway. There were 4 DBDs that were not validated. None of the 11 GIDs were validated. The 4 non-validated DBDs were Radiation Monitoring, Main Steam, Condensate System, and Feedwater. As discussed in Attachment G, CP&L is committing to complete the validation, in accordance with the DBD program plan, of the DBDs and GIDs that were not validated previously. This action will be completed by October 15, 1998.

A DBD discrepancy resolution process was implemented based upon the requirements in CP&L procedure DBD.PII.1, "Discrepancy Resolution," Revision 0 and Revision 1, to resolve discrepancies identified during the DBD development or the DBD validation process. This procedure provided for screening and prioritization of discrepancies based on risk significance.[4] The procedure also invoked CP&L's Corrective Action Program (CAP) for discrepancies that were categorized as adverse conditions.

Some of the problems found during the DBD development and validation process were deemed less significant at that time and did not qualify as DBD discrepancies. As discussed in Attachment G, CP&L is committing to resolve these less significant open items by December 15, 1997

4 The risk categories were: 1) Recommend Plant Shutdown, 2) Immediate Evaluation, 3) Place in Integrated Schedule, and 4) Potential Drop (i.e., no follow-up actions).

III. Brunswick Nuclear Plant Design Basis Documents

A. *General Description*

The Brunswick Nuclear Plant (BNP) "Design Basis Reconstitution Project" was conducted based on guidance in a documented CP&L Project Plan that provided the instructions for activities associated with the compilation of design bases information for BNP systems and generic issues. There were six main elements of the project specified in the Project Plan:

1) Establishment of a hierarchy of design information;
2) Design bases information collection and consolidation;
3) Engineering evaluation and development of Design Basis Documents (DBDs) for selected systems and topics;
4) Validation of the information in the DBDs;
5) Discrepancy resolution; and
6) Control of the DBDs.

These six elements were consistent with industry guidance on design bases programs that were under development at the time (i.e., NUMARC 90-12, "Design Basis Program Guidelines") and the Design Basis Reconstitution Project that had been undertaken at the Robinson Nuclear Plant. However, two significant reconstitution efforts, related to electrical distribution and piping design, were initiated and completed for BNP. These efforts primarily involved performing new calculations.

The DBD format developed for BNP was based on the lessons learned from the Robinson Nuclear Plant (RNP) DBD effort, the Harris Nuclear Plant (HNP) DBD effort, and the guidance in NUMARC 90-12, "Design Basis Program Guidelines." The BNP project was accomplished at a time when the industry had settled on a consistent set of guidelines for conduct of DBD projects. The primary purpose of the BNP project was to structure design bases information and supporting calculations/analyses applicable to plant systems and generic issues into DBDs, and control them for future use.

The DBDs were compiled, reviewed, and approved by CP&L, with input from the BNP Nuclear Steam Supply System (NSSS) vendor (General Electric) and the BNP Architect/Engineer (United Engineers and Constructors). The systems for which DBDs would be compiled were selected based on whether the system was required for safe shutdown or accident mitigation. A total of 27 system DBDs and 12 generic issue DBDs were created for BNP.

B. *Program Scope*

The 27 system DBDs are as follows:

DBD-01	Nuclear Boiler System
DBD-02	Reactor Recirculation System
DBD-03	Reactor Protection System
DBD-04	Pressure Suppression System
DBD-05	Standby Liquid Control System
DBD-08	Control Rod Drive Hydraulic System
DBD-09	Neutron Monitoring System
DBD-10	Standby Gas Treatment System
DBD-11	Radiation Monitoring System
DBD-12	Primary Containment Isolation System
DBD-17	Residual Heat Removal System
DBD-18	Core Spray System
DBD-19	High Pressure Coolant Injection System
DBD-20	Automatic Depressurization System
DBD-23	Containment Penetration System
DBD-24	Containment Atmospheric Control System
DBD-37	Control Building HVAC System
DBD-37.1	Reactor Building Ventilation System
DBD-37.4	Diesel Generator Building Ventilation Air
DBD-38	Fuel Oil System (for Emergency Diesel Generators)
DBD-39	Emergency Diesel Generator System
DBD-43	Service Water System
DBD-46	Instrument and Service Air System
DBD-50	AC Electrical System
DBD-51	DC Electrical System
DBD-58	Structures Systems
DBD-60	Emergency Response Facility Information System/Safety Parameter Display System

The 12 generic issues DBDs are as follows:

DBD-100	Environmental Qualification
DBD-101	Appendix R (Fire Protection)
DBD-102	Seismic Qualification
DBD-103	ALARA (As Low As Reasonably Achievable)
DBD-104	Radiation Shielding
DBD-105	Postulated Pipe Failure Analysis
DBD-106	Hazards Analysis
DBD-107	Regulatory Guide 1.97 (Instrumentation)
DBD-109	Human Factors
DBD-110	Single Failure Criteria
DBD-111	Station Blackout
DBD-112	Cable and Raceway

C. DBD Correctness and Accessibility

The BNP DBDs are based on referenced source documents that are listed throughout the DBDs and that are generally maintained as records within the Quality Assurance Program records control process. The DBDs are controlled documents that are available in the plant and corporate libraries.

D. Maintaining DBDs Current

The BNP DBDs are considered "design documents," and as such they are required, by CP&L procedures, to be maintained in a controlled format that is useable by CP&L organizations. Changes to DBDs are tracked by the Nuclear Revision Control System (NRCS), which is CP&L's configuration management tool for design documents. NRCS allows for maintenance of the current status of design documents and control of outstanding changes in a consistent and timely fashion. Nuclear Engineering Department procedures require that the DBDs be considered during preparation of plant modifications/design document changes, and that they be revised if the plant modification/design document change impacts the design bases information in the affected DBDs. Also, CP&L's Engineering Support Personnel (ESP) Training Program provides training on DBD use, and the Corrective Action Program (CAP) provides the mechanism for documenting, investigating, and correcting conditions adverse to quality, including identified discrepancies between plant configuration or operation and the Updated/Final Safety Analysis Reports (U/FSARs).

E. Program History

In order to provide a systematic and comprehensive approach to the research, development, and writing of the BNP system and generic issues DBDs, the DBD preparation was performed in accordance with procedure NED 3022, "BNP DBD Writer's Procedure." This procedure was consistent with NUMARC 90-12 in addressing the "what" and "why" of a system's design bases (i.e., "what" refers to the actual design bases information and "why" refers to the supporting design information that provides the rationale for the design bases[5]). It was also consistent with the format set in the RNP DBD project. The format for system DBDs includes the system functional requirements, regulatory imposed design requirements and commitments, system design requirements, component design requirements, design margin, and references to applicable design bases documents.

In order to provide reasonable assurance that the system design, operation, and surveillance testing was consistent with the critical design basis parameters,[6]

5 NUMARC 90-12, "Design Basis Program Guidelines" dated October 1990.

6 "Critical design basis parameters" were not specifically defined in procedure NED 3021, but refer to the 10CFR50.2 definition for design basis, i.e., specific values or ranges of values chosen for controlling parameters as reference bounds for design of a system that performs a specific safety-related function.

requirements for DBD validation and discrepancy resolution were provided in procedure NED 3021, "BNP DBD Validation and Discrepancy Resolution," for the 27 safety-related system DBDs.[7] This procedure included provisions for validation review of applicable design descriptive documents, calculations, operating procedures, surveillance procedures, and various other documents/data bases for the DBD system. The procedure also provided for screening and prioritization of discrepancies based on risk significance[8] and invoked CP&L's Corrective Action Program (CAP) for discrepancies that were categorized as adverse conditions.

Plant Management presented the BNP Design Basis Reconstitution Project Plan to NRC during a meeting at NRC's Region II office on December 3, 1990. Since that time, NRC has reviewed the BNP Design Basis Reconstitution Program and documented its findings in various NRC Inspection Reports (e.g., NRC Inspection Report Nos. 50-325/92-10 and 50-324/92-10, dated April 29, 1992; NRC Inspection Report Nos. 50-325/92-39 and 50-324/92-39, dated November 20, 1992; and NRC Inspection Report Nos. 50-325/94-20 and 50-324/94-20).

7 NOTE: Only safety-related system DBDs were validated (i.e., Generic issue DBDs were not validated).

8 The risk categories were based on the Probabilistic Risk Assessment (PRA) core damage frequency associated with failure:

1) Very High Risk or greater than 10^{-4}/year,
2) High Risk or greater than 10^{-6}/year to 10^{-4}/year,
3) Moderate Risk or greater than 10^{-7}/year to 10^{-6}/year,
4) Low Risk or less than 10^{-7}/year, and
5) Not Applicable.

IV. Harris Nuclear Plant Design Basis Documents

A. *General Description*

The Harris Nuclear Plant (HNP) received a full-power operating license in 1987. HNP required the Architect/Engineer (Ebasco) to create Design Basis Documents (DBDs) as part of the turnover of design documents from Ebasco to CP&L as plant construction was completed and prior to plant commercial operation. This effort was undertaken prior to the industry guidelines for DBDs being established (i.e., NUMARC 90-12, "Design Basis Program Guidelines") and was CP&L's earliest effort to create such a set of documents.

The HNP DBDs were created in the 1985-1987 time frame. A set of 65 DBDs are currently approved and maintained. No specific design bases information reconstitution effort resulted from the creation of the HNP DBDs since the original design was being completed at the same time that the DBDs were being prepared.

CP&L completed receipt of the HNP DBDs from Ebasco in 1987 and incorporated them as part of the turnover of design documents from Ebasco to CP&L.

B. Program Scope

The 65 DBDs currently maintained are:

DBD-001	Pipe Supports
DBD-002	Piping Stress Analysis
DBD-003	HVAC Ductwork
DBD-004	Seismic Monitoring System
DBD-018	Containment Building Platform Design
DBD-100	Reactor Coolant System (RCS)
DBD-101	RCS and Secondary Sampling System
DBD-102	Reactor Makeup Water System
DBD-103	Chemical and Volume Control, Boron Thermal Regeneration, Boron Recycle
DBD-104	Safety Injection System
DBD-105	Residual Heat Removal System
DBD-106	Containment Spray System
DBD-108	Containment Isolation
DBD-109	Fuel Handling System
DBD-110	Fuel Pool Cooling and Cleanup System
DBD-111	Metal Impact Monitoring System (i.e., loose parts monitoring)
DBD-112	Condensate, Main Feedwater, Condensate Polishers, Feedwater Drains and Vents
DBD-113	Circulating Water, Condenser, Cooling Tower, Circulating Water Treatment
DBD-114	Auxiliary Feedwater
DBD-115	Secondary Waste Treatment
DBD-116	Filter Backwash System
DBD-117	Spent Resin Storage and Transfer System
DBD-118	Floor Drain Storage and Treatment System
DBD-119	Waste Holdup and Evaporation System
DBD-200	Cable and Raceway System
DBD-120	Laundry and Hot Shower System
DBD-121	Radioactive Sampling System
DBD-122	Gaseous Waste Processing System
DBD-123	Steam Generator Wet Layup
DBD-124	Oily Waste Collection and Separation
DBD-125	Steam Generator, Main Steam, Extraction Steam, Steam Dump, and Auxiliary Steam
DBD-126	Steam Generator Blowdown and Chemical Addition System
DBD-127	Main Turbine, Turbine Generator Lube Oil, Gland Seal and Exhaust Hood Spray, Moisture Separator Reheater, Digital Electro-Hydraulic, and Hydrogen Seal Oil Systems
DBD-128	Service Water System
DBD-129	Water Treatment System
DBD-130	Caustic and Acid System
DBD-131	Component Cooling Water System
DBD-132	Essential and Non-Essential Chilled Water
DBD-133	Compressed Air System
DBD-134	Bulk Gas Storage
DBD-135	Fuel Handling Building and Waste Processing Building HVAC
DBD-136	Containment Ventilation and Cooling System
DBD-137	Reactor Auxiliary Building HVAC System
DBD-138	Control Room HVAC
DBD-139	Turbine Generator Building HVAC
DBD-140	Diesel Generator Building HVAC System
DBD-141	Plumbing and Drainage System

DBD-201	Emergency Diesel Generator System
DBD-202	Plant Electrical Distribution System
DBD-203	Plant Lighting System
DBD-204	Station Grounding and Cathodic Protection
DBD-205	Load Frequency Control System
DBD-206	Plant Communications Systems
DBD-301	Reactor Control and Protection System
DBD-302	Control Rod Drive and Rod Cooling Systems
DBD-303	Excore/Incore Nuclear Instrumentation System
DBD-304	Radiation Monitoring System
DBD-305	Post Accident Hydrogen Analyzer System
DBD-306	Fire Protection and Detection System
DBD-307	Emergency Response Facility Information System and Waste Processing Monitoring and Display
DBD-308	Main/Auxiliary Control Boards and Panels
DBD-309	Integrated Security Systems
DBD-310	Post Accident Sampling (Reactor Coolant)
DBD-313	Time Response
DBD-HNT-131-01	Hot Machine Shop

The system DBDs typically included:

1) The system functional requirements;
2) The design bases and assumptions;
3) The design margin; and
4) A document reference list.

The original HNP effort did not include development of generic safety issue DBDs.

C. *DBD Correctness and Accessibility*

The HNP DBDs are based on referenced source documents that are listed throughout the DBDs and that are generally maintained as records within the Quality Assurance Program records control process. The DBDs are controlled documents that are available in the plant and corporate libraries.

D. *Maintaining DBDs Current*

The HNP DBDs are considered "design documents," and as such they are required, by procedure, to be maintained in a controlled format which is useable by CP&L organizations. Changes to DBDs are tracked by the Nuclear Revision Control System (NRCS), which is CP&L's configuration management tool for design documents. NRCS allows for maintenance of the current status of design documents and control of outstanding changes in a consistent and timely fashion. Nuclear Engineering Department procedures require that the DBDs be considered during preparation of plant modifications/design document changes and that they be revised if the plant

modification/design document change impacts the design bases information in the affected DBDs. Also, CP&L's Engineering Support Personnel (ESP) Training Program provides training on DBD use, and the Corrective Action Program (CAP) provides the mechanism for documenting, investigating, and correcting conditions adverse to quality, including identified discrepancies between plant configuration or operation and the Updated/Final Safety Analysis Reports (U/FSARs).

E. *Program History*

As described earlier, HNP DBDs were created as part of the original plant design documents by the Architect/ Engineer such that a validation, discrepancy resolution, and/or reconstitution program was not needed. As a result, HNP received an operating license with an existing set of DBDs.

Attachment G

Compilation of Commitments and Planned Enhancements

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I. Introduction

This attachment provides a consolidated summary of CP&L's planned actions to provide additional confidence that the design bases are appropriately translated into the design and operation, maintenance and testing procedures, and that the configuration and performance of systems, structures, and components remain consistent with the design bases.

The planned actions, which are discussed in the prior attachments, are categorized as either commitments or enhancements. Commitments are made (a) in response to specific NRC requests, or (b) at CP&L's initiative to assure compliance with NRC regulations and/or previous CP&L commitments. Enhancements are those additional actions CP&L is planning to take to provide further assurance that design bases are appropriately translated into the design and operation, maintenance and testing procedures and that the configuration and performance of systems, structures, and components remain consistent with the design bases. These actions are considered enhancements because they are not considered necessary to bring CP&L into compliance with NRC regulations. The schedule for completion of these enhancements will be developed and maintained at the respective sites.

II. Actions Applicable to BNP, HNP and RNP

A. Commitments:

1) Complete the FSAR Improvement Initiative as described in the May 30, 1996 presentation to the NRC based upon the schedule to be maintained at the individual sites.
2) Complete the NEI Initiative described in accordance with the NEI letter from Mr. J. F. Colvin to the Honorable Shirley A. Jackson dated August 2, 1996.
3) Until NRC questions concerning the Configuration Change (CC) process are resolved, procedure EGR-NGGC-0005 requires that safety related Configuration Change (CC) ESRs be treated as Design Change (DC) ESRs (i.e., a Design Verification Review is required to be performed).

B. Enhancements:

1) Complete an assessment of configuration management. Include in this assessment an evaluation of potential actions to further enhance the accessibility and retrievability of licensing and design bases information and linkages between procedures. The results of this assessment will be used to determine additional actions required.
2) Provide additional training on licensing and design bases to (a) procedure technical reviewers, (b) personnel performing unreviewed safety question determinations per 10CFR50.59, and (c) engineers performing design modifications. This enhanced training will be incorporated into the routine training programs.
3) Perform self-assessments on CP&L's effectiveness in design bases translation into procedures and configuration maintenance in accordance with Section 17.3 of the U/FSARs. Include in these self-assessments, conducted in accordance with site procedures, appropriate vertical and horizontal slices of systems and programs. Consider design bases information in these assessments. The specific assessments to be performed will be determined based upon the findings from the corrective action program, the operating experience programs and management judgment.

III. Actions Applicable to BNP

A. *Commitments:*

1) Perform an evaluation of the verification and validation practices for outsourced engineering work. This commitment is consistent with the action identified in CP&L letter BSEP 96-0449 to NRC dated December 23, 1996.
2) Revisit the UFSAR and DBDs to ensure that the containment analysis inputs and acceptance criteria are clearly defined. This commitment is consistent with the action identified in CP&L letter BSEP 96-0449 to NRC dated December 23, 1996.

   (Note: In addition to Commitments (1) and (2) above, CP&L further committed in BSEP 96-0449 to factor the lessons learned from the power uprate into the evaluations performed in the preparation of this response. This has been completed.)

3) Install the ECCS Suction modification per NRC Bulletin 96-03.
4) Complete implementation of the Improved Technical Specifications (ITS) after NRC approval of the previously submitted BNP application. The implementation and associate procedure revisions provide an opportunity to enhance our confidence that the design bases are properly translated into plant procedures.
5) Complete the actions requested in Generic Letter 96-01.
6) Complete the walkdowns and corrective actions in response to noted deficiencies in the Environmental Qualification Program.

B. *Enhancements:*

1) Review selected designs of safety-related modifications that were prepared in the time period 1992-1995. One purpose of this review will be to verify appropriate consideration was given to design bases information and the effectiveness of its translation into plant procedures and other documents.
2) Review selected technical programs to verify appropriate consideration was given to design bases information and its translation into these programs, as well as ensuring appropriate procedural guidance and accountability is defined.
3) Complete the power uprate for Units 1 and 2. The power uprate process has facilitated a thorough review of associated analyses for proper consideration of the design bases.
4) Incorporate any outstanding revision requests in the current project being performed to bring maintenance procedures into compliance with the administrative controls procedure described in Attachment B, as well as

perform validation of procedures where required. This project is due for completion by September 1, 1997.

5) Resolve outliers (items not meeting the evaluation criteria as set forth in the GIP and requiring further analytical evaluation) by the end of the Unit 1 refueling outage scheduled for the spring of 1998.
6) Complete resolution of Design Basis Document (DBD) discrepancies currently being tracked in the Corrective Action Program.

IV. Actions Applicable to HNP

A. *Commitments:*

1) Submit, for NRC approval, an application to convert to the ITS. Complete implementation of the ITS after approval of the application by the NRC. The implementation and associated procedure revisions provide an opportunity to enhance our confidence that the design bases are properly translated into plant procedures.
2) Per the response to NOV 96-01-01, a plant modification will be installed that provides alarming capabilities for RABEES boundary doors.

B. *Enhancements:*

1) Use the steam generator replacement and power uprate project planned in the near future as an opportunity to provide additional confidence in SSC configuration and performance. (As part of these projects, many of the design calculations will be reviewed and revised.)
2) Define the role of the design bases documents (DBD) within the HNP configuration management program and then assess and upgrade the DBDs as necessary to meet that role.
3) Track the necessary procedure revisions identified in the self-assessment of procedure revisions resulting from modifications and addressed in Engineering Service Request (ESR) 95-01041 to completion.
4) Review the issues identified in the self-assessment report for the Safety System Functional Evaluation of the Reactor Trip System and develop corrective actions.
5) Implement the two modifications to correct design deficiencies described in the Improvement Initiatives Section of Attachment B under "Review of Surveillance Testing of Safety-Related Logic Circuits."
6) Implement the revisions of additional procedures described in the Improvement Initiatives Section of Attachment B under "Review of Surveillance Testing of Safety-Related Logic Circuits."
7) Resolve conditions that have been identified as a result of the follow-on assessment involving the potential inappropriate use of the clearance process through the Corrective Action Program following NOV 96-11-06.

V. Actions Applicable to RNP

A. *Commitments:*

1) Complete implementation of the ITS after NRC approval of the previously submitted RNP application. The implementation and associate procedure revisions provide an opportunity to enhance our confidence that the design bases are properly translated into plant procedures.
2) Complete validations, in accordance with the DBD program plan, of the DBDs and the GIDs that were not validated previously as part of the Design Basis Reconstitution Program. This action will be completed by October 15, 1998.
3) Discrepancies identified as part of the Design Basis Reconstitution Program were evaluated and dispositioned during the program. Certain less significant items identified during the review are in the ESR action item tracking system and are assigned to System Engineers for resolution. Priority will be increased on resolving these open items so that they are resolved by December 15, 1997.