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SUBJECT: Responds to RAI re adequacy & availability of design bases info.

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SUSQUEHANNA STEAM ELECTRIC STATION
RESPONSE TO NRC REQUEST FOR INFORMATION
REGARDING ADEQUACY AND AVAILABILITY OF
DESIGN BASES INFORMATION — **Docket Nos. 50-387 and 50-388**
PLA-4546 — **FILE R41-2**

References: 1) Letter from J.M. Taylor (NRC) to W.F. Hecht (PP&L), "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," dated October 9, 1996.

The objective of this letter is to respond to the subject NRC request for information (Reference No. 1). The purpose of the request is to obtain information that will provide the NRC with added confidence and assurance that Susquehanna Steam Electric Station (SSES) Units 1 and 2 are operated and maintained within the design bases, and that any deviations are reconciled in a timely manner.

We are aware that recent NRC industry inspections have found that design basis information has not been appropriately maintained and implemented at certain plants. Although we have confidence in our management systems that assure the adequacy and availability of design basis information, we also recognize the need to evaluate these issues. Corporate management is committed to this effort. We have closely monitored industry events, and in early 1996, initiated a proactive effort to evaluate the applicability of such findings to PP&L. We will continue to monitor the effectiveness of our design and configuration controls and will take actions to enhance our processes, when appropriate. We are also cognizant of the challenges inherent in evaluating the original design bases using state-of-the-art analytical tools.

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Requested Information

Specifically, the NRC requested the following information:

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

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

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

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

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

The referenced letter further requests an indication as to whether PP&L has undertaken any design review or reconstitution programs, and if not, a rationale for not implementing such a program.

Our detailed response to your request for information is contained in the attachment to this letter, which is organized into six major sections. In addition to the overview provided in this letter, Section I of the attachment provides an introduction to the general approach taken in responding to your request. Responses to the five items listed above are provided in Sections II though VI. Our response to your additional request regarding design review or reconstitution activities is contained within this letter.

Overview of PP&L Response

Our response is essentially presented in terms of the original design basis "baseline" and our management of change since establishing the baseline. The response describes the processes used to translate the design bases into the physical plant configuration and plant procedures, as well as feedback mechanisms which provide us with confidence that these processes have been, and are, effective and that plant configuration, performance and procedures conform with design bases.

Our response addresses three primary points, as listed below.

(1) Design Baseline

Recognition of the rigor of the start-up process is an important consideration, and a key to our confidence that fidelity has been maintained between the design bases and the plant configuration. Specifically, the turnover and start-up testing activities provided reasonable assurance that the original design was consistent with technical standards, that the design basis information provided a sufficient foundation from which to control changes, and that the plant configuration reflected the design basis information. Consistency between design bases and the plant configuration was also confirmed through an independent design verification of the feedwater system in support of initial plant licensing.

A quality assurance (QA) program was developed to apply the criteria of 10 CFR 50, Appendix B to the design and construction phases of the SSES project. Incorporation of QA program requirements into work practices and procedures, coupled with independent oversight of these practices and procedures, provided a QA program framework for design and configuration control.

PP&L established early ownership of the design and configuration of SSES through extensive turnover and start-up testing activities. The significant involvement of PP&L's Integrated Start-Up Group throughout the start-up process established an understanding of the importance of consistent translation of design bases into plant procedures and the plant configuration. PP&L's definition and execution of an extensive engineering turnover process provided additional assurance that the design bases information necessary to maintain the plant and its design margins was available. PP&L's early involvement in the development of plant procedures, in parallel with the start-up testing, also helped to assure that the plant design, as translated into start-up testing, was also translated into plant procedures. PP&L's participation in turnover walkdowns and initial testing established an understanding of the importance of consistent translation of the design bases into the physical plant configuration, as well as an understanding of the value of testing as a means of confirming consistency between operating system performance and design bases.

(2) Current Processes for Design, Configuration and Deficiency Controls

From this design and configuration baseline, PP&L has maintained ownership of the design, operation and maintenance of the plant using qualified and experienced personnel, and appropriate procedures. The knowledge-base of our personnel has continued to grow, in turn allowing PP&L to retain ownership of design, rather than relying heavily upon contractors for design work.

Additionally, the QA program framework upon which initial processes were developed, was expanded upon by a set of Supplemental Procedures to define and facilitate the transition between construction and operational phases of SSES, and ultimately by the development of the PP&L Operational QA Program. Incorporation of QA program requirements into operational activities (e.g., operations, testing, maintenance, modification, etc.), along with continuing independent oversight, provide a framework for design and configuration control.

Within the framework of the QA program, our processes have continued to evolve. PP&L recognized the need to make the transition between the design processes utilized during the design and construction phases to a set of controls more appropriate for changes made during the operational phase. PP&L's current integrated set of design and configuration control processes include mechanisms for considering the design bases during design, transforming a design change into information describing the configuration of the plant, and then finally for implementing design changes into the actual physical configuration of the plant.

Processes have been established to assure that as the plant is modified, procedures are reviewed by knowledgeable personnel and are appropriately updated to translate the design change. In addition to periodic reviews to determine whether changes are necessary, operating, maintenance and testing procedures are reviewed, and revised as necessary, following events in which the procedure contributed to the cause of the event or was inadequate in mitigating the effects of the event. Configuration management requirements are an integral part of the plant modification program. The modification process itself is designed to assure that the design bases are identified, are properly converted into the design, are reflected in design output documents and then into the actual physical plant configuration. For example, the process used to install and close-out Design Change Packages (DCPs) includes development of a Plant Modification Package by engineering and plant personnel to identify all work documents, procedure changes and testing requirements necessary to implement the modification.

Our mechanisms for problem identification and corrective action are used to identify anomalies in the design and configuration controls (i.e., "translation" processes) described above. Implementation of problem identification and corrective action processes allows us to correct inconsistencies between the design bases and the plant configuration, performance and procedures, and to prevent recurrence of these problems through feedback to the design and configuration control processes. This assures that processes evolve through lessons learned from operating experiences.

(3) Feedback on Process Effectiveness

PP&L has undertaken major projects which provided unique opportunities for feedback on the effectiveness of design and configuration control processes used to translate design bases. Additionally, audits and assessments (both internal and external) have provided evidence supporting our confidence in the effectiveness of our processes and programs.

(a) Power Uprate: One recent example of a major modification is the "uprate" of both SSES units in 1994 (Unit 2) and 1995 (Unit 1) to a higher licensed thermal power level. Part of the Power Uprate Program included studies to review each affected plant system (Nuclear Steam Supply Systems and balance-of-plant systems) to determine the system's ability to support the uprated conditions. This included review of the original design requirements and bases for each system, the current actual plant operating conditions, and the conditions which would be expected to exist at the uprated power condition. At the end of the Power Uprate Program implementation refueling and inspection outage, each unit was returned to power operations through the implementation of an extensive test program, which encompassed activities from verification of the newly configured reactor core to performance and evaluation of testing at the new power level. This test program was conducted with considerations for tests and administrative controls similar to those used during the original Start-up Test Program.

(b) Improved Technical Specifications (ITS): Another broad initiative which is responsive to the issues raised in your request is our recent submittal of Improved Technical Specifications (ITS) for NRC review and approval. Technical Specifications, as Appendix A to the Operating License, set forth the limits, operating conditions, and other requirements imposed upon facility operation for the protection of the health and safety of the public. Given this scope, development of Improved Technical Specifications (ITS) for each SSES unit required an extensive reevaluation of existing design and licensing documentation, as well as implementing procedures. The SSES ITS Project, initiated in April of 1995, essentially reevaluated these bases for system operation. The ITS Project began with the development of initial review packages for each section of the SSES ITS, and cross-functional reviews throughout the Department (e.g., Operations, Chemistry, Health Physics, Nuclear Engineering, Licensing, etc.). During the development of the initial review package, the SSES current Technical Specifications were *not* used as a design source. Instead, controlled sources of design and licensing basis information were used to develop the SSES ITS. Design limits specified in the SSES ITS were verified, as were statements in the SSES ITS Bases. If a design limit or statement could not be verified via consistency with an approved design document, an SSES ITS open item was

created and issued to the appropriate PP&L organization for disposition. This project served to reconfirm the consistency among licensing, design, and "operating" bases, for safety significant systems within the scope of ITS. A submittal to the NRC was made in August 1996.

(c) Audits and Assessments: Continuing oversight by PP&L's independent QA function, along with other inspections and assessments have provided feedback on process implementation. Note that in some cases, these reviews have identified various weaknesses related to control of the design bases or consistency between the design bases and the physical plant. Upon identification, these items are entered into the corrective action process where they are evaluated for safety significance, operability, and reportability, and where appropriate, corrective actions are implemented. Examples of "verification" activities include the following:

- A Safety System Functional Inspection (SSFI) performed by PP&L on the Emergency Service Water System in 1988.
- An Electrical Distribution System Functional Inspection (EDSFI) performed by the NRC in 1990.
- An SSFI performed by an independent team on the High Pressure Coolant Injection System in 1992.
- A Service Water System SWOPI Readiness Inspection performed by the NRC in 1995.

The relative lack of safety significance associated with the few deficiencies identified during these activities provides added confidence that design bases have been translated into plant procedures and configuration. This conclusion is also consistent with results to date from our Design Basis Documentation (DBD) Project described below.

Ongoing PP&L Actions

(1) Design Basis Documentation (DBD) Project: In response to your question regarding whether PP&L has embarked upon a design review or reconstitution program, PP&L began a design basis initiative, referred to as the DBD Project, in 1992 that is primarily directed at better organizing its design basis information. Because the initial design basis information for the SSES units was technically sound and well controlled, complete design basis reconstitution was unnecessary for SSES. However, when our design basis

initiative (or other assessment, change or corrective action processes) has indicated the need for reconsideration in specific areas, PP&L has taken, and will continue to take, the necessary corrective actions. The DBD Project has generally followed the "mixed approach" presented in NUMARC 90-12. The mixed approach uses a combination of text and extensive references to document design bases. Forty-nine DBDs have been selected for development, on the basis of criteria including: safety-related systems, NSSS systems, risk-significant systems as defined by the maintenance rule, important balance-of-plant systems, customer needs, etc. To date, a total of 18 DBDs have been developed. Additionally, approximately 23,000 calculations, which effectively represent the design calculations for SSES, have been scanned into an optical disk storage system and re-indexed for easy retrieval by personnel via a Department information system.

(2) Current Licensing Basis (CLB) Project: As part of the overall Nuclear Department Assessment Plan developed in 1995, PP&L initiated an assessment effort in February 1996 to assess the SSES FSAR relative to emerging industry issues. The general purpose of the CLB Project is to assure the overall health of PP&L's current licensing basis. Specific objectives include: characterizing the "health" of the FSAR and taking immediate action, where warranted; focusing the problem evaluation such that enhancement actions can be effectively and efficiently executed; and identifying and resolving any potential programmatic/process weaknesses. The initial assessment revealed that a number of apparent FSAR discrepancies had been identified via DBD Open Items (see discussion above), and remained outstanding. Additionally, a number of Licensing Document Change Notices (LDCNs) required expedited processing. Specific recommendations for process improvements were also identified.

The project commenced in June 1996, and as of this writing, is ongoing. The two primary areas of emphasis include: executing near-term actions to address the recommendations from the initial assessment; and performance of a broad "scoping" assessment of the FSAR. To date, "apparent" FSAR discrepancies identified in the initial assessment have been dispositioned, most requiring no action (i.e., no actual discrepancy existed) or simple FSAR updates. None of the deficiencies identified to date have been safety significant. Nevertheless, we will perform an aggregate analysis of identified discrepancies in order to identify any generic process implications. Additionally, recommended process improvements are in development and LDCN incorporation was expedited to ensure all updates were incorporated.

Scoping reviews of the FSAR are continuing, as are FSAR update activities. The final phase of the CLB Project is not anticipated to be finalized until mid-1997 to ensure that all scope inputs are considered, including the issues identified in the NRC October 9, 1996 request for design basis information. Specific "feeders" to the scope of the final phase of the project scope include: (1) the results of findings from the aggregate analysis and scoping reviews referenced above; (2) recommendations from the team charged with development of the response to your October 9, 1996 request; and (3) results of implementing NEI Initiative 96-05. These feeders will include process improvements, short-term and long-term actions and recommendations for additional assessment.

Conclusion

We have reasonable assurance that design bases requirements are consistent with the plant procedures, configuration and performance. This is based in part on our confidence in the processes themselves, as well as on feedback which provides evidence of effective process implementation in translating the original design, and changes to the design over the operating history of SSES.

Our reasonable assurance is also based upon our confidence in our quality management systems. Our QA program has provided the framework upon which we have developed and refined our design "translation" processes, throughout the design, construction and operational phases of SSES. This program and supporting processes have evolved over the years, due in part to our QA oversight and assessment functions. Additionally, we have accrued benefits from the expanding knowledge-base of our personnel over time, in implementing these processes.

In summary, our reasonable assurance that the design bases have been translated into plant procedures, configuration and performance is based upon the following key points:

- Establishment of a solid original design baseline, using qualified personnel.

- Implementation of integrated change management and corrective action processes, using qualified personnel.

- Feedback on the effectiveness of the processes in translating design bases through continuing activities such as oversight and assessment (internal and external), as well as through unique opportunities provided via major projects.

We will continue to look for deficiencies and process improvement opportunities related to the issues in this letter with our ongoing CLB Project.

If you have any questions or require any additional information, please contact Mr. James Kenny at (610) 774-7535.

Very truly yours,

R. G. Byram

Attachment

copy: NRC Region I

Mr. K. Jenison,	NRC Sr. Resident Inspector
Mr. C. Poslusny,	NRC Sr. Project Manager
Mr. H. Miller,	NRC Regional Administrator - Region I
Mr. S. Collins,	Director, Office of Nuclear Reactor Regulation

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA)
: SS
COUNTY OF LEHIGH)

I, ROBERT G. BYRAM, being duly sworn according to law, state that I am Senior Vice President - Nuclear of Pennsylvania Power & Light Company and that the facts set forth on the attached response regarding the adequacy and availability of design basis information are true and correct to the best of my knowledge, information and belief.

Robert G. Byram
Senior Vice President - Nuclear

Sworn to and subscribed before me this 13th day of February, 1997.

Martha C. Sedora
Notary Public

Notarial Seal
Martha C. Sedora, Notary Public
Allentown, Lehigh County
My Commission Expires Jan. 15, 1998
Member, Pennsylvania Association of Notaries

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ATTACHMENT TO PLA-4546

TABLE OF CONTENTS

I. Introduction

A. Purpose

The purpose of this document is to respond to the "Request for Information Pursuant to 50.54(f) Regarding Adequacy and Availability of Design Bases Information," dated October 9, 1996 by the NRC, and received October 21, 1996 by PP&L. The intent of this response is to provide information to give the NRC added confidence and assurance that the Susquehanna Steam Electric Station (SSES) is operated and maintained within the plant design bases, and that deviations are reconciled in a timely manner. For the reasons described herein, PP&L concludes that there is reasonable assurance that the plant configuration, performance and procedures are consistent with the design bases.

Evaluation of the consistency of design bases with actual plant configuration, performance and procedures is an ongoing process. We anticipate that we will continue to find discrepancies from time to time. As described in Section V of this response, PP&L will evaluate apparent inconsistencies and disposition these items as they are identified. We are also cognizant of the challenges inherent in evaluating original design bases using state-of-the-art analytical tools which have evolved since the development of the original design bases.

The process descriptions provided in this response are based upon current versions of Nuclear Department procedures. It is expected that continuing improvement efforts such as reviews, audits and self-assessment (e.g., the ongoing CLB Project described in Section VI) will identify opportunities for process enhancement. As a result, processes will evolve as a normal course of business through controlled procedural changes. This response summarizes PP&L's existing programs and processes, and is not intended to identify new licensing commitments.

B. Overview

Our response is essentially presented in terms of the original design basis "baseline" and our management of change since establishing the baseline. The response describes the processes used to translate the design bases into the plant configuration and plant procedures, as well as feedback

mechanisms (i.e., verification activities) which provide us with confidence that these processes have been, and are, effective.

The discussion which follows summarizes PP&L's response, section-by-section.

- Section II responds to item (a) of the NRC request, describing those processes used by PP&L to establish the design bases, both originally and through plant changes. In addition to providing a description of how the original design bases was established through engineering turnover, and confirmed through start-up testing and an independent design verification of the feedwater system, this section describes the quality assurance process framework within which start-up processes were conducted. Section II also describes PP&L's current integrated set of design and configuration control processes. Also established within the framework of the quality assurance program, these processes include mechanisms for considering the design bases during design, transforming a design change into information describing the configuration of the plant, and then finally for implementing design changes into the actual configuration of the plant.

- Section III responds to item (b) of the NRC request, addressing the question of how design bases were originally translated into plant procedures, and how design changes are translated into plant procedures. Our rationale for concluding that there is reasonable assurance that the design bases requirements are translated into operating, maintenance and testing procedures, is based in part on our confidence in the processes themselves, as well as on verification activities which provide evidence of procedure adequacy.

- Section IV responds to item (c) of the NRC request, and discusses the processes used to translate the original design into the plant configuration, as well as the current processes used for this purpose. Again, our rationale for concluding that there is reasonable assurance that design bases requirements are consistent with the plant configuration and system performance is based in part on our confidence in the processes themselves, as well as on verification activities which provide evidence of acceptable plant configuration and performance.

- Section V responds to item (d) of the NRC request, and addresses our mechanisms for problem identification and corrective action. These are the processes we use to identify anomalies in the design and configuration controls (i.e., "translation" processes) described in

Sections II through IV, and in the plant configuration, performance and procedures. Effective implementation of our problem identification and corrective action processes allows us to correct inconsistencies between the design bases and the plant, and to prevent recurrence of these problems with feedback to the design and configuration control processes. This ensures that processes evolve through lessons learned from operating experiences.

- Finally, in response to item (e) of the NRC request, Section VI provides a summary of the overall effectiveness of our processes and programs in ensuring consistency between the design bases and the plant configuration. This response segment is derived primarily from Sections II through V, and reiterates three key points. These include: (1) our solid "baseline"; (2) our integrated set of processes for managing changes to the baseline; and (3) feedback on process effectiveness through the incorporation of quality assurance principles in the way we do work. This last point is supported by descriptions of quality assurance audit and assessment (internal and external) activities which evaluated design and configuration controls; major projects which provided comprehensive reviews of the plant; and the ongoing self-assessment initiative initiated in early 1996 to identify discrepancies and opportunities for process improvement.

C. Organization

This response follows the organization of the NRC request for information. Five major response segments follow this Introduction section, to parallel the five specific areas addressed in your request.

II. Response to Item (a): Description of Design and Configuration Control Processes

A. Introduction

1. Purpose

This section responds to the following requested information:

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

2. Overview

From the earliest stages of design and construction of the Susquehanna Steam Electric Station (SSES), PP&L assumed responsibility for, and provided an active role in, overseeing these activities. PP&L established and implemented a Quality Assurance (QA) program for the design and construction phases that was responsive to 10 CFR 50, Appendix B. PP&L responsibility was identified through design, construction and start-up testing.

When the time arrived for PP&L to assume direct responsibility for the design and operation of SSES, the QA program was expanded by a set of Supplemental Procedures (SPs) that defined the transition between the construction and operational phases of SSES. The SPs were handled as amendments to the PP&L QA Manual which developed into the Operational QA Program, as defined in the Final Safety Analysis Report (FSAR), Chapter 17.2, and the Operational Policy Statements (OPS). Key activities addressed within the SPs were such QA practices as design control; nuclear fuel management; transfer and control of plant materials, equipment, structures and systems; control of plant maintenance; and inspection of pre-operational testing. QA requirements governed the various phases of this transition from design and construction to operation, including construction

turnover, engineering turnover and initial testing. These transitional activities were designed to provide reasonable assurance that PP&L had assumed responsibility for a quality product; had reasonable assurance that the plant was built and performed consistent with the design bases; and had sufficient design information to define the design bases and margins to allow the plant to be safely maintained, operated, and modified consistent with the design bases. Our engineering personnel gained the knowledge necessary to understand the design bases.

Since that time, PP&L has maintained and expanded design, configuration management, and document control processes to assure integrated processes for managing changes to the original design baseline. Effective application of quality assurance principles coupled with the solid original design baseline provides added confidence that PP&L has appropriately managed the design and configuration of SSES.

3. Organization

This section provides a summary description of the original design, construction, testing and turnover processes, and a description of the current processes established to manage changes to the configuration of SSES. The detailed response to item (a) is divided into the following four sections:

- Establishment of the Original Design Bases
- Current Framework for Design and Configuration Control
- Description of Design Control Processes
- Description of Configuration Control Processes

B. Establishment of the Original Design Bases

1. PP&L Framework for Design, Construction and Start-Up

PP&L assumed the responsibility to ensure that SSES Units 1 and 2 were designed, constructed and tested prior to initial operation, in accordance with applicable regulations, codes, and specifications. To fulfill this responsibility, PP&L established a project organization comprised of engineers, technicians, operators, and QA/QC personnel to maintain continual involvement and control

throughout the project by overseeing the design, construction, and testing phases of SSES.

This program was described in Appendix D to the Preliminary Safety Analysis Report (PSAR) and was documented in the Susquehanna QA Program (SQAP). The SQAP scope applied throughout the phases of design, procurement, manufacturing and fabrication, construction and installation, and preoperational testing. In addition to its application to those structures, components and systems that prevent or mitigate the consequences of a postulated accident which may cause undue risk to the health and safety of the public, the SQAP extended to those structures, components and systems that provide for unit reliability, availability, and maintainability consistent with PP&L system requirements.

The PP&L QA Program described in Appendix D of the PSAR, delineated the scope, policies, practices, and principles followed by PP&L and the principal contractors associated with the original design and construction of SSES. Principal contractors included: (1) Bechtel Corporation (Bechtel), the architect/engineer and builder of the plant; and (2) General Electric Company (GE), the supplier of the Nuclear Steam Supply System (NSSS). Bechtel interfaced with GE to assure coordination of the original design. PP&L ensured that work by these contractors, as well as several of their major subcontractors, was executed in accordance with appropriate quality assurance controls delineated within written QA programs. PP&L activities included: in-line reviews of design documents and changes thereto; participation in the resolution of design and construction problems; witnessing of factory testing of equipment prior to shipment to the plant for installation; QA and engineering audits and surveillances of activities performed under the various QA programs; participation in system walkdowns; and witnessing of testing associated with system tuning.

During the construction phase, the design of the plant evolved, and the provisions of the QA Program were designed to assure that the physical plant conformed to the latest revision of the design output documents. Assurance that the current design met technical standards was provided throughout this phase of the project, so that future work could build upon the current design. Engineering personnel from PP&L provided oversight of the design work performed by both Bechtel and GE. Personnel from PP&L's construction group provided oversight of the construction

activities. PP&L's QA group provided the direction for the Bechtel and GE quality programs.

PP&L sent many engineers to interface directly with Bechtel and GE engineers, as a means of providing input, and gaining first-hand SSES design knowledge. Design areas in which PP&L engineers had extensive involvement include: advanced control room, stress corrosion cracking, electrical voltage drop analysis, quencher testing with Kraftwerk Union, environmental qualification, and seismic modeling.

PP&L also initiated an effort to develop the in-house capability to support core reload design and licensing activities. The first step in this process was the development of three-dimensional simulation models of the SSES initial reactor core designs and plant system models of the SSES units. The results from these models were compared to calculated data from the FSAR to assess model accuracy (the initial design of the SSES reactor core is documented in GE report "NEDO-20944-P, BWR/4 and BWR/5 Fuel Design" and was incorporated in the FSAR via reference). These models were then used to perform pre-start-up predictions in support of the start-up and test phases. The measured data obtained during the start-up and test phases were used to perform post-start-up evaluations of many of the start-up tests. Based on the results of the post-start-up evaluations, the models were further improved to more accurately reflect plant measurements. In addition to the modeling of the SSES Units, PP&L also developed models of other nuclear plants to enable further comparisons to measured data. As a result of the efforts to develop PP&L's 3-D simulation models and plant system models, PP&L developed the in-house knowledge and expertise necessary to support the current in-house reload design and licensing capabilities.

During the design and construction phases of the project, PP&L became involved with many design activities. For example, in response to ASME code requirements, many design bases requirements at the time of construction were issued through System Design Specifications (SDS) which provided very specific design parameters for each SSES safety system. ASME Class 1 systems were certified, by professional engineers in the Class 1 Stress Reports, to be designed in accordance with the requirements of the applicable System Design Specifications. In addition, ASME piping systems, including pipe supports, have been as-built in accordance with NRC IE Bulletin 79-14, "As-Built Safety Related Piping Systems," and the as-built information has been

included and/or reconciled in the final calculations and reports. The progression of the design process for piping systems, from the original System Design Specifications, to the certification provides confidence in design bases consistency in the piping area.

PP&L formed the Integrated Start-up Group (ISG) which reported directly to the Plant Superintendent, to establish and manage the preoperational test program. The ISG was composed of personnel representing various engineering disciplines from PP&L, Bechtel, and GE and interfaced with plant maintenance, instrumentation, and operations personnel who supported performance of the testing. The operating group was established through an extensive recruitment process which assured qualified and experienced operators were utilized during the testing phase of the project. These personnel were responsible for the initial power ascension testing and subsequent operation of the units. Using qualified personnel, PP&L established the start-up, operating, and maintenance control process. PP&L QA/QC personnel conducted inspections and monitoring of testing activities.

Prior to start-up, PP&L recognized the need to reconcile the design bases, design output, and configuration of the physical plant. This led to the early development of system-level scoping drawings and component databases which identified design output documentation. Following turnover, PP&L incorporated configuration controls and the Design Change Process (DCP) to facilitate management of further design and configuration changes. These controls relied upon the concept of "as engineered vs. as built" documentation. (Information sources maintained in an as-engineered condition are provided with a change mechanism that assures a level of review and approval suitable to the specific information source, and establishes the information update requirements so that the updated information is available upon approval. Information sources maintained in an as-built condition are controlled such that the as-engineered condition is shown and tracked, and the as-built condition is shown only after the actual change has been made.) This early approach assured that PP&L developed the SSES design and configuration in an orderly fashion.

2. Construction Turnover

As the construction of systems/components was completed, the construction organizations transferred jurisdictional control over these systems/components through a formal turnover evolution with PP&L. This turnover process was formally established in procedures under the PP&L QA program and entailed detailed walkdowns by Bechtel and PP&L construction engineers, Bechtel and PP&L QA/QC, and the PP&L ISG to establish both construction status and conformance of the physical plant with design documentation. As with other activities performed under the quality assurance principles defined in the PP&L QA Manual, the performance of these turnovers and resolution of issues were subject to QA audits and surveillances.

Identified issues were tracked in a punch-list to ensure resolution. PP&L engineering, plant staff, and QA/QC personnel actively participated in the resolution of the these issues. QC inspections of physical work necessary to resolve the open issues were performed to assure the adequacy of the physical plant in conforming to the design documents.

PP&L's ISG maintained system jurisdiction during the preoperational test program until the release of the systems from test status. Subsequently, design change controls were applied to limit the scope of the change, identify the affected systems, and assemble all the drawing change mechanisms in one package.

3. Initial Test Program

The Initial Test Program commenced with system/component turnover and terminated with the completion of power ascension testing. As described in FSAR Section 14.2, the program was conducted to confirm that design parameters were: (1) within required ranges, and (2) consistent with the design bases submitted in support of the application for the SSES Operating License. This testing was performed in accordance with written procedures, which were prepared, reviewed and approved in accordance with quality principles that conformed to 10 CFR 50, Appendix B Criteria V, VI and XI. The test procedures incorporated the requirements and acceptance limits contained in applicable design

documents and were developed with input from SSES plant staff and engineering personnel.

As described below, testing during the Initial Test Program was accomplished in distinct and sequential phases.

a) *Phase I: Component Inspection and Testing*

Component inspection and testing ensured that: (1) components and equipment were calibrated and checked, (2) construction work on a particular system was completed to the degree required, and (3) the system was initially operated and prepared for subsequent testing. Control of component testing was performed in accordance with plant start-up procedures. These procedures established a method for administratively controlling the performance of equipment or component testing, test data review, test data reconciliation, and documentation development. The procedures detailed the objectives, acceptance criteria, references (including Bechtel and/or vendor documentation), prerequisites, precautions and notes, test equipment, procedural steps, figures, tables, and appendices.

QA and QC personnel provided independent audit, surveillance, and inspection functions as a means of assessing the adequacy and effectiveness of these testing activities in assuring satisfaction with established acceptance criteria and management expectations. These QA activities were performed in accordance with 10 CFR 50, Appendix B, Criteria X and XVIII.

b) *Phase II: Preoperational and Acceptance Testing*

Preoperational tests demonstrated, to the extent practicable, the capability of safety-related structures, systems, and components to meet their safety-related performance requirements. Preoperational testing was controlled in accordance with the preoperational/acceptance test procedures. The procedures were developed, reviewed and approved in accordance with quality assurance controls that conformed to 10 CFR 50, Appendix B Criterion V, VI and

XI. The procedures detailed the objectives, acceptance criteria, references, prerequisites, precautions and notes, test equipment, system test steps, inspection hold points, and appendices. To the extent practicable, the objectives of the testing included:

- Verifying the adequacy of plant design;
- Verifying that plant construction was in accordance with design;
- Demonstrating proper system/component response to anticipated transients and postulated accidents;
- Confirming the adequacy of plant operating and emergency procedures; and
- Familiarizing plant staff operating, technical, and maintenance personnel with plant systems.

Test acceptance criteria identified those measures necessary to determine that system performance was acceptable. These criteria were quantitative or qualitative. Quantitative acceptance criteria specified system or equipment parameters in accordance with design values of process variables and equipment operating characteristics (e.g., flows, temperatures, pressures, currents, voltages, etc.) required under specific conditions. Qualitative acceptance criteria specified system or equipment design functions, (e.g., automatic start, sequencing, or shutdown) occurring under specified conditions. The performance of the testing was subject to QC inspection and monitoring as defined by the QA program.

Test procedures also listed the documents and revisions used to support procedure preparation. Examples included: FSAR references, applicable Regulatory Guides, logic diagrams, flow diagrams, single-line diagrams, single-line meter and relay diagrams, piping and instrumentation diagrams, electrical schematics diagrams, instrument indices, material requisitions or specifications, and vendor data.

c) *Start-Up Testing*

Phase III: Initial Fuel Loading
Phase IV: Initial Heat-Up and Low Power Testing
Phase V: Power Ascension Testing

The start-up test program commenced with the start of nuclear fuel loading and terminated with the completion of power ascension testing and the warranty run. The procedures were developed, reviewed, and approved in accordance with quality assurance principles that conformed to 10 CFR 50, Appendix B, Criteria V, VI, and XI. The GE Operations Manager reviewed and approved acceptance criteria and test objectives of NSSS start-up tests. Bechtel project engineering reviewed and approved acceptance criteria and test objectives of non-NSSS start-up tests. The acceptance criteria were used to determine that system performance was acceptable. The procedures detailed test objectives, test descriptions, acceptance criteria and sources of each acceptance criterion, references, prerequisites, precautions, test equipment, procedural steps, and appendices (included forms, figures, tables, or supplementary material). PP&L plant staff with support from engineering personnel reviewed and approved the start-up test procedures in addition to participating in the performance of the tests.

Start-up test procedures referenced those documents used during procedure preparation and included: FSAR references, material requisitions and/or specifications, Technical Specifications, operating procedures, regulatory guides, piping and instrument diagrams, electrical schematic drawings, start-up test specifications, and start-up test instructions.

These tests confirmed the system/functional design bases identified in the SAR, and demonstrated, to the extent practicable, that the plant operated in accordance with design and was capable of responding as designed to anticipated transients and postulated accidents. Testing was sequenced such that the safety of the plant was never totally dependent upon the performance of un-tested structures, systems, and components. The objectives of the start-up test program included:

- Accomplishing a controlled, orderly, and safe initial core loading;
- Accomplishing a controlled, orderly, and safe initial criticality and heat-up;

- Conducting low-power testing sufficient to ensure that design parameters were satisfied and safety analysis assumptions were correct or conservative; and
- Performing a controlled, orderly, and safe power ascension.

In addition to the conduct of periodic QA audits of the program, in-line QA reviews of test results were performed in order to provide for an independent check as to the completeness and adequacy of the testing program.

4. Engineering Turnover

As stated earlier, PP&L engineering personnel were involved throughout the design construction, and testing phases of the project to assure that processes were properly controlled and that design bases were understood. In addition, PP&L defined an extensive engineering turnover process to assure its ability to control the design bases and the configuration of the as-engineered vs. as-built plant. This turnover process with the Architect/Engineer (Bechtel) was another step in assuring that the design bases information necessary to maintain the plant and design margins was available for the PP&L engineers, plant staff and training personnel. Certain information could not be turned over due to its proprietary nature. We obtained necessary design bases information on an as-needed basis.

The purpose of the engineering turnover was to provide the information necessary to define the plant design bases and configuration in sufficient detail, such that analyses and design modifications could be properly managed by PP&L. In order to meet this objective, engineering turnover consisted of a package of information for each system which contained the following:

a) Design Criteria

The Design Criteria documented the design requirements for the system in sufficient detail to obtain structure-level, system-level and major component-level design inputs for analysis and modification design. The Design Criteria included the following types of information for the system as a whole, and for major components:

- References to licensing documents (e.g., FSAR, Environmental Report, Security Plan, Fire Protection Plan, Emergency Plan, etc.).

- References to applicable NRC requirements and guidance, including an explanation of the extent of conformance where appropriate.

- References to applicable industry codes and standards (e.g., IEEE, ANSI, ASME, and ASCE), and an explanation of the extent of conformance if appropriate.

- Key operating parameters/requirements which had to be met for the system to perform its intended function (e.g., flow rates, heat loads, data scan rates, communications channel-band widths, man-machine interfaces, etc.).

- System operating modes for which the system operating parameters must be met.

- Seismic and environmental requirements for the system and key components.

- System functional requirements, which defined the functions to be performed by the computer system.

b) Functional Assurance

Functional Assurance provided the documentary record that the Design Criteria were accurately translated into the completed design. It consisted of references to QA records, test results, and calculations to evidence that Design Criteria had been successfully met. The Functional Assurance was included in the QA/QC records for the project turned over to PP&L, with some retained by Bechtel and GE.

c) *Configuration Documentation*

Configuration Documentation performed the dual function of defining the design and providing the mechanism by which change could be documented for release to the field. Configuration Documentation was included in the Engineering Turnover (ETO) Packages. The drawings, flowcharts, source codes and specifications were accurate descriptions of the completed systems.

d) *Design Evolution*

The Design Evolution provided the historical record of the design development. It included references to studies, letters, meeting minutes, and calculations to answer the question "why?" when the choice was not dictated by Design Criteria or test result. Design Evolution was included in the project documentation lists and the QA/QC records.

The format of the Engineering Turnover Packages included the following elements:

- Section I - Drawings: Lists of scoped drawings obtained from Bechtel's configuration control group, sorted by start-up system. For computer systems, the scoped drawing list was obtained from the computer system vendor.

- Section II - Calculations: List of calculations applicable to the system and a copy of any in final status. For computer systems, this section contained a list of program design specifications and program module listings.

- Section III - Components: Component indices obtained from Bechtel's configuration control group, sorted by start-up system. For computer systems, the component indices were obtained from the computer system vendor.

- Section IV - Exceptions: Revision 0 itemized the outstanding work required to complete the system.

- Section V - Description: A brief description of the ETO Package.

PP&L's acceptance of the engineering turnover information included a sampling review to ensure: (1) that adequate information was provided to completely define the system; (2) that the information contained in the documentation was accurate and applicable to the system, structure or facility being defined; and (3) that outstanding engineering work items were identified, and the requirements for completion were defined and scheduled.

Acceptance ensured that the design was adequately defined so that the document could be used as a design reference source. PP&L QA personnel conducted audits and surveillances of the turnover process to assure implementation of the established process and closure of any issues identified during the turnover.

In addition, nuclear engineering, plant staff and nuclear training personnel performed a review of system descriptions developed for PP&L. After engineering turnover, design changes for the portion of the plant turned over to PP&L were released for installation such that PP&L became the control point for both design control and configuration control. This avoided the risk of "competing" designs being issued without coordination.

5. Initial Licensing Activities

PP&L had extensive, detailed interactions with the NRC on the design of SSES in support of issuance of the Operating License. PP&L applied for the SSES Operating Licenses on April 10, 1978, and the Unit 1 license was issued July 17, 1982. During this period, numerous meetings took place among the NRC, PP&L, Bechtel and GE. The FSAR typically served as the focal point for the discussions, documenting both NRC questions and the subsequent agreements reached as a result of the meetings. Over 1000 of the questions, on all facets of the design and operation, are documented in the final three volumes of the nineteen volume FSAR. In addition to these questions by the NRC technical review staff, many design issues were reviewed during ASLB and ACRS hearings.

A significant factor in the length of time it took to obtain the initial operating license was the TMI accident in 1979. The NRC developed an extensive action plan (see NUREGs 0660, 0694, and 0737) that PP&L was required to respond to in order to be licensed. PP&L's response is documented in Chapter 18 of the FSAR, and NRC's acceptance is documented as part of their Safety Evaluation Report for SSES (NUREG-0776, including seven supplements).

Another key milestone during initial licensing occurred in March, 1982, just prior to licensing of SSES Unit 1, when design errors were revealed during the licensing of another nuclear facility. In response to these concerns, the NRC requested additional assurance that Unit 1 had been properly designed and constructed, i.e., in accordance with the application. This request was met through the performance of an independent design verification of the mechanical and structural design of the Feedwater system. The major elements of this verification, performed by an independent contractor, included reviews of design control, interface control, document control, control of field changes, nonconformance and corrective actions, audit findings, installation inspections, and as-built documentation. Some findings regarding the adequacy of piping supports were identified and corrected. Overall, this effort provided substantial confirmation that SSES met the requirements of its application. For Unit 2, licensed June 27, 1984, PP&L was not required to perform this review, based upon the following:

- Submittal of documentation of audits and reviews that confirmed design activities;
- Submittal of a self-initiated evaluation of the Unit 2 construction project using INPO's "Performance Objectives and Criteria for Project Evaluations"; and
- Similarity of the design process to Unit 1, and incorporation of improvements based on the Unit 1 independent design verification process.

In summary, the scope and diligence of the licensing process is a significant reason that PP&L has confidence that the original plant design basis was well-founded.

C. Current Framework for Design and Configuration Control

PP&L established programs for design control and configuration control to meet the applicable requirements of 10 CFR 50, Appendix B. The processes described in this section are current procedures which have been updated since the time of licensing. The procedures have incorporated revisions due to organizational changes, new requirements, and improvements developed through audits, assessments, and inspections. The preceding section described information and factors considered to establish the plant design basis through the transition of design and construction from Bechtel and GE to PP&L. The process used to control the development of the design and the construction of the plant provided the basis to make an orderly transition to operations. This section describes the integrated process PP&L is using to control changes to the plant design. Application of these controls coupled with the solid design bases established at start-up provides added confidence that PP&L has appropriately controlled the design and configuration of SSES. The assurance provided by the quality audits and assessments, and other activities, strengthens this confidence.

1. Personnel

Each functional unit manager is responsible for training and certifying or qualifying their personnel to ensure that they can competently, safely, and efficiently perform their assigned duties. The Training Department provides a link to the balance of the Department to assure that other affected personnel (including operations and maintenance) get the training necessary to keep abreast of the system level changes. Training has been an integral part of communicating the plant design changes and procedural changes to the organization.

Since PP&L became the control point for design and configuration control of SSES, many issues have been identified and resolved. Engineering personnel have continued to expand their knowledge of the design bases through involvement on projects and corrective actions. In engineering, training has been an integral part of communicating the plant design and procedural changes to the PP&L organization. The Engineering Support Personnel Training has specified qualification training required to perform the engineering duties. As an example, within the last two years, the engineering training has addressed: licensing basis documentation, 10 CFR 50.59 refresher training, Technical Specification training, and systems level training. The Engineering Support Personnel Training provides a communication tool to keep personnel updated on changes, and also represents an additional link to the balance of the Department to assure that other affected personnel get the training necessary to keep abreast of the system-level changes.

2. Procedure Hierarchy

PP&L's set of approved procedural controls represent a systematic approach to assuring adequate engineering design and configuration control processes, including those that implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR 50. The discussion below describes engineering design and configuration control processes to illustrate the integrated set of controls used to control the design and configuration of SSES, as well as the manner in which they "tier" from NRC regulations. PP&L has established engineering design and configuration controls in formalized programs and procedures. The requirements upon which these programs and procedures are founded are defined within the PP&L Operational Quality Assurance (OQA) Program described in FSAR Chapter 17.2, which is responsive to 10 CFR 50, Appendix B.

The following discussion describes PP&L's framework for compliance with the requirements of 10 CFR 50, Appendix B, 10 CFR 50.59, and 10 CFR 50.71(e).

a) 10 CFR 50, Appendix B

In response to 10 CFR 50, Appendix B, PP&L established requirements for assuring the quality of safety-related activities during the operations phase. The requirements include:

- FSAR Chapter 17.2, which contains PP&L's quality assurance program;
- the OQA Manual, which contains 18 OPS which define "upper-tier" requirements for general application to PP&L activities within the scope of the OQA Program,

and expand upon the broad requirements through the inclusion of 10 CFR 50.59 and other specific requirements contained in FSAR Table 17.2-1;

- Nuclear Department Administrative Procedures (NDAPs), which define department-wide programs, processes and controls; assign responsibilities to individual functional units; and delegate authority for the operation of the Nuclear Department; and

- Functional Unit Procedures (FUPs), which contain the detailed information necessary for each organizational unit (i.e., "functional unit") within the Nuclear Department to comply with the OPS and NDAPs. The relationships between these documents are shown in FSAR Figure 17.2-1.

Since the design and configuration control activities are under the scope of the OQA Program, the programs and processes are subject to periodic formal audits established to comply with 10 CFR 50, Appendix B Criterion XVIII. These formal audits are supplemented by self-assessments and independent assessments as a means of evaluating the adequacy and effectiveness of the programs and processes in providing the required end results.

b) *10 CFR 50.59*

The 10 CFR 50.59 process begins once the Operating License, which reflects the basis for NRC approval of the design and operation of the facility, is issued. This regulation exists to ensure that proposed changes which impact the basis for the Operating License are not implemented without prior NRC review and approval. PP&L's program for implementing 10 CFR 50.59 has two key elements: (1) screening questions to determine if a proposed change falls within the scope of 10 CFR 50.59 (i.e., a "50.59 determination"), and (2) the actual questions that must be answered to determine whether or not an unreviewed safety question exists (i.e., a "safety evaluation"). If an unreviewed safety question is identified, PP&L must submit a proposed amendment to the Operating License pursuant to 10 CFR 50.90 and receive an approved

amendment from the NRC prior to implementing the change. 10 CFR 50.4 requires the periodic submittal to the NRC of a summary of safety evaluations performed under the provisions of 10 CFR 50.59.

PP&L performance of 10 CFR 50.59 safety evaluations has evolved considerably since initial operation. Early procedures that basically reiterated the regulatory requirements have been supplemented by significant guidance that has originated from several sources over time. These are listed below.

- Training, beginning with external experts, and currently by PP&L trainers, has been provided to evaluators, reviewers, and approvers. Continuing engineering training is currently providing refresher training that among other aspects, examines PP&L CLB Project findings (see Section VI) and recent industry problems in implementing 10 CFR 50.59.

- Modifications process improvements have provided focused efforts to ensure that safety evaluations consider pertinent potential impacts through the proceduralization of a "design considerations list." This list has been, and will continue to be, expanded to include specific areas, as a result of corrective action reviews and lessons learned.

- Industry efforts to develop NSAC 125, "Guidelines for 10 CFR 50.59 Safety Evaluations," dated June 1989, also contributes to process evolution. PP&L recognizes that the NRC has documented a short list of disagreements with this document, but overall, PP&L believes that the effort resulted in improved understanding and consistency of application of 10 CFR 50.59.

- Oversight functions, including internal QA, PORC (Plant Operations Review Committee), ERC (Engineering Review Committee), and SRC (Susquehanna Review Committee) monitor performance and provide feedback. (In recent years, the latter formed a subcommittee devoted specifically to reviewing 10 CFR 50.59 Safety Evaluations.)

PP&L procedures include the process for preparing a 50.59 determination, explanation of when to write a Safety Evaluation, scope of the written Safety Evaluation, general considerations of the Safety Evaluation, details of the written Safety Evaluation, results of the Safety Evaluation, and record requirements. Engineers that prepare, review, and approve Safety Evaluations must have the required training in accordance with the Engineering Support Personnel Training Program. Safety Evaluations are prepared and approved by engineering and submitted to the PORC for review and recommendation of approval to proceed with the modification. The release of the work packages to the field implementation groups is dependent on the PORC approval. The engineer responsible for the modification normally makes the presentation of the Safety Evaluation to PORC. If a revision to the modification is required that impacts the Safety Evaluation, then a revised Safety Evaluation must be prepared to address the impact of the revision. Again, PORC must approve the revised Safety Evaluation in order for the revision to be released to the field.

Recent ongoing efforts to review the adequacy of PP&L's maintenance of the FSAR have resulted in a number of proposed 10 CFR 50.59 process enhancements that are currently under review. Examples include: the need to clarify and reinforce the definition of the SAR to facilitate user reviews of applicable licensing documents beyond the FSAR and Technical Specifications; the value of providing better documentation of the 50.59 screening determination that a proposed change does not fall within the scope of 10 CFR 50.59; the need to enhance the linkages between 10 CFR 50.59 evaluations and the licensing document update process; and the need to enhance the linkages between 10 CFR 50.59 and similar regulatory reviews, such as those performed in accordance with 10 CFR 50.54, or under the provisions of the Environmental Protection Plan.

Based upon the above, PP&L believes that its 10 CFR 50.59 implementation has been monitored, and improved over the operating history of SSES. Recent efforts will create further refinements, which in conjunction with training and information system upgrades allowing greater

ease of access to SAR documents, will serve to ensure improved implementation.

c) *10 CFR 50.71(e)*

This regulation requires licensees to update the FSAR to assure that it contains up-to-date information, such that both the licensee and the regulator have a consistent, accurate view of that portion of the licensing basis required to be provided by the FSAR. The FSAR is part of the application for the Operating License, and changes to it are required to be processed under the provisions of 10 CFR 50.59. 10 CFR 50.71(e) then serves to ensure that these changes are properly documented in the FSAR.

The SSES FSAR consists of 19 volumes, and was developed in accordance with Regulatory Guide 1.70, Revision 2. PP&L issued Revision 0 of the FSAR in 1978. Since that time, it has been updated 50 times, approximately 20 of which have occurred since issuance of the Unit 1 Operating License in July, 1982.

Examples of process improvements which have occurred since initial operation include efforts to enhance understanding of 10CFR50.59, as well as a number of efforts aimed at helping the user, e.g., providing internal updates more frequently than required by 10 CFR 50.71; and a database of pending FSAR changes accessible by desktop PC by personnel to provide quick access to other changes that could impact their work. In addition, major projects such as Design Basis Documentation (DBD) Project and the Power Uprate Program (PUP) have provided wide-ranging input to the FSAR that has improved the quality of its content.

PP&L's process for updating the FSAR is part of a procedure for controlling changes to a number of licensing documents. The key attributes of the process related to the FSAR include:

- Review to identify the need for an FSAR change as part of various plant change mechanisms;

- 10 CFR 50.59 review, including establishing the requirement for prior NRC approval due to identification of the need for a change to the Technical Specifications;
- Multi-disciplined change review by affected department functional units, including a quality assurance review;
- Administrative controls that ensure proper release of individual changes that are held in abeyance pending other actions, such as a designed modification becoming operational, or NRC issuance of an approved license amendment;
- Record keeping in accordance with regulatory requirements; and
- Submitting FSAR updates to the NRC in accordance with 10 CFR 50.71(e).

In early 1996, efforts were initiated by PP&L to review the adequacy of PP&L's maintenance of the FSAR as part of a broader assessment of the SSES Current Licensing Basis (CLB). Our initial reviews identified a number of minor discrepancies (many identified during the DBD Project) that indicated the need for further work. As part of the CLB Project (see Section VI), PP&L is currently performing an in-depth review of the FSAR, including vertical slice reviews, to further examine its quality. To date, no safety significant concerns have been identified in broad "scoping" reviews by department subject matter experts. However, certain FSAR sections have been found to contain outdated information, and efforts are underway to correct those problems, understand their root causes, and provide effective process improvements that will prevent recurrence.

D. Description of Design Control Processes

The requirements upon which SSES design controls are founded are defined within the PP&L OQA program. The design control process utilized by PP&L is in compliance with 10CFR50, Appendix B, Criterion III. The program is implemented by experienced engineering personnel. Although, PP&L has confidence in the program as it exists today in meeting regulatory requirements, it is also recognized that the process of improvement will be continual through the operating phase of SSES.

1. Overview of Design Control Processes

PP&L defines "design" as the technical and management processes that commence with identification of design input, and which lead to, and include, the issuance of design output documents. As described above, the information provided through the design, construction, and testing evolution provided a solid foundation from which PP&L established the engineering design and configuration control processes to assure continual control of the plant design and bases. PP&L has maintained the design bases by applying systematic processes with the use of qualified personnel to control plant changes. The processes described below illustrate PP&L's systematic approach to controlling plant changes and maintaining consistency with the design bases. Compliance with the regulations is integral to the design process (i.e., engineers follow 10 CFR 50.59 and 10 CFR 50.71(e) requirements because they are incorporated as part of the design process).

A key element of the design process is the use of Design Standards and Design Guides, which have been developed to communicate uniform design criteria and methods. The supplemental methods and procedures specified in Design Standards help to assure a consistent, technically-acceptable engineering design. Design Guides provide job aids and reference material.

A change to the plant design requires either a change in design inputs or formal calculation and/or analysis to establish that the design inputs are met. Applicable design inputs (e.g., regulatory requirements, data tables, codes and standards, and design bases) are reflected in design output documents (e.g., specifications, drawings). These design output documents and the controlling procedures specify the appropriate quality standards.

a) Identification of Design Inputs

The design process begins with the definition of design inputs, which establish the criteria, parameters, bases or other design requirements upon which the design is based. Design inputs establish the uniform set of criteria which governs all disciplines in the design process. They are the criteria which must be reviewed and revised as changes or adjustments are made throughout the detailed design and field support phases of the design process.

The procedure describing design inputs and considerations is utilized to identify those criteria, parameters, bases, or other design requirements upon which the detailed final design is based. Certain technical areas, due to their complexity or breadth of technical scope, are given special programmatic attention in "design considerations." The primary purpose of the design considerations is to assure that proper engineering involvement and/or design program interfaces occur. The concept of design inputs originates in ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants." FSAR Section 17.2.3 and Table 17.2-1 (with noted exception) provide PP&L's commitment to compliance with Regulatory Guide 1.64, which endorses ANSI N45.2.11. Operational Policy Statements (OPS) require compliance with the requirements listed above.

The development of design inputs provides a necessary tool for the engineer in the design process. Applicable design inputs, such as design bases, regulatory requirements, codes and standards, and user-identified requirements are identified, documented, and their selection reviewed and approved. Changes to design inputs, including the reason for the changes, are identified, approved, documented and controlled. Design inputs are specified to the level of detail necessary to permit the design activity to be carried out in a correct manner, and to provide a consistent basis for making design decisions, accomplishing design verification and/or validation measures, evaluating design changes and formulating the functional requirements document or procurement specification.

Tools to allow electronic access to a variety of key design and licensing bases information sources have been developed. The Screen Managed Automated Retrieval Technology System (SMARTS) has been developed to provide users with text-searchable electronic copies of DBDs and selected licensing documents. Hyperlinking to electronic images of most DBD references is also provided. This information is available via desktop computers throughout the Department.

b) Selection of Materials and Equipment

Processes which control selection of materials and equipment meet the requirements of 10 CFR 50, Appendix B, Criteria III, IV, and VII. Based upon the approved design inputs, the design engineer evaluates and selects suitable materials, parts, equipment, and processes for safety-related systems, structures, and components.

This evaluation and selection includes the application of appropriate industry standards and specifications. Materials, parts, and equipment which are standard, commercial, or which have been previously approved for a different application, are reviewed for suitability in the intended application prior to use. A new procurement specification may be prepared for the equipment, if required by the complexity of the procurement. Alternatively, a more concise technical data specification may be used to specify limiting and/or overall criteria (e.g., performance, sizing, and/or material requirements) or to impose requirements on a vendor's standard product design when the extent and complexity of the requirements do not justify the development of a procurement specification.

The engineer's selection of materials and equipment relies upon PP&L systems for material control, which are comprised of supplier evaluation, audits, source verification, receipt inspection, and evaluation of supplier records. The extent and methods of control used assure compliance with applicable technical, manufacturing, and quality requirements. Following receipt of the equipment and prior to installation, PP&L's QC receipt processes assure that equipment received on site meets the quality requirements of the specification. Vendor testing and functional testing following installation assure that the equipment meets design bases requirements.

c) *Calculations and Analyses*

Because calculations are used to document technical decisions, establish design bases and demonstrate compliance with codes, standards, criteria and design bases, they constitute a fundamental building block in the design process. PP&L's processes that control engineering calculations and studies meet the requirements of 10 CFR 50, Appendix B, Criteria III, V, and VI. Preparation of an engineering calculation or study is performed in accordance with procedural requirements. These requirements establish a standard method for assignment of identification numbers, development of review and approval requirements, and records retention and retrieval.

Significant improvements in the calculation indexing and retrieval processes have occurred in the last several years. Approximately 23,000 calculations, which effectively represent the design calculations for SSES, have been scanned into an optical disk storage system and re-indexed for easy retrieval via the SMARTS information system described above. (Note: 4000 of the very large piping and civil calculations only have summary sheets scanned in to refer the user to hard copy calculations.)

d) *Design Review and Verification*

Designs are reviewed in accordance with QA procedures that conform to 10 CFR 50, Appendix B Criterion III to assure that design characteristics are verified and acceptance criteria are identified. Design documents produced by, or for, Nuclear Engineering undergo reviews by qualified individuals. This is to assure the correctness of the inputs, assumptions, engineering thought-processes, conclusions, and products of the design, regardless of the quality classification of the design activity, in accordance with procedural guidance.

Safety-related designs are verified by independent review, alternate calculations, or qualification testing. Design verification is performed by a qualified individual or group

other than the original designer or the designer's immediate supervisor. (Note: Supervisors may perform design verification subject to the restrictions of Paragraph C.2 of Regulatory Guide 1.64, Revision 2 as modified by FSAR Table 17.2-1.) The verifier reviews the design document to ensure that: (1) the content is responsive to the request made, (2) required inputs have been identified and approved, and (3) the design approach used, and the documents produced, appear technically acceptable and complete. Methods utilized to perform verifications are specified in procedures and include the items listed below.

- Design reviews are performed by a qualified engineer, and either use a detailed line-by-line checking of the design document or a multiple group review with the verifier performing the review coordination.

- Alternate calculations provide a comparison with alternate methods of calculation or analysis. These address the appropriateness of assumptions, input data, and the code or other calculational method used.

- Qualification testing demonstrates the adequacy of performance under the most adverse design conditions. Pertinent operating modes are considered in determining those design conditions where it is intended that the test program confirm the adequacy of the overall design. The qualification testing is performed in accordance with written test procedures. These incorporate or reference the requirements and acceptance limits contained in the applicable design documents. Qualification Testing conforms to 10CFR50, Appendix B, Criteria XI and XIV.

e) ***Design Output Documentation***

Procedures are established to control the flow of design output information between organizations. These procedures include methods for the review, approval, release, distribution, and revision of documents involving design interfaces with other organizations. The design output documentation, in conjunction with the use of qualified and experienced personnel, procedures, training,

and QA/QC audits and assessments, provide an effective mechanism to ensure adequate communication of design bases information to affected organizations.

Preparation of specifications is performed in accordance with written procedures. Specifications are design output documents intended to define and communicate technical requirements, usually to other functional organizations, both within and external to PP&L. Appropriate subjects for specifications include: procurement requirements, installation requirements, inspection and test requirements, welding requirements, ASME Code design requirements, and technical data.

Drawings are one of the primary methods of documenting design output requirements created by calculations and analyses. Drawings are a source of plant configuration information for operations, maintenance, and engineering personnel. Drawings are created in accordance with procedural requirements.

The Nuclear Information Management System (NIMS) is an evolving electronic database application that will, when completed, contain much of the work management processes and administrative information used to support Nuclear Department activities. NIMS procedures describe the responsibilities associated with the system. NIMS currently contains technical information about the plant equipment. When completed, it will also contain much of the operations and maintenance information. As such, it will become a major repository of information, eventually containing electronic drawings and other technical documents. As operations, maintenance and other functions are supported by NIMS, design requirements data will be available directly, instead of solely through intermediate hard copy.

As a direct result of the evolution of the use of computers, the Software Quality Assurance Program (SQAP) has been established to apply the 10 CFR 50, Appendix B requirements. This program is contained in Appendix OPS-1-A to the OQA Manual which defines the scope of the program and the applicable QA elements. This program is based on industry guidance and practices, such as ASME NQA-2a-1990, Part 2.7 "Quality Assurance

Requirements for Computer Software for Nuclear Facility Applications"; IEEE 7.4.3.2-1993, "Application Criteria For Programmable Digital Computer Systems In Safety Systems Of Nuclear Power Generating Stations"; IEEE Standard 730-1981, "Software Quality Assurance Plans"; and NUREG/CR-4640, "Handbook Of Software QA Techniques Applicable To The Nuclear Industry." Other Nuclear Department procedures implement this program.

Licensing documentation (e.g., the FSAR) is updated by using a Licensing Document Change Notice (LDCN). This is performed in accordance with the procedure for controlling changes to licensing documents.

Design Basis Documentation (described in more detail in Section VI) is updated by a Design Basis Document Change Notice (DBDCN) in accordance with the Design Basis Documentation procedure. The DBDCN is created, reviewed and approved, and either issued directly or included in a modification package for inclusion, when the modification is installed.

The Environmental Qualification Data Base (EQDB) is a computer information system which provides a source of select Environmental Qualification (EQ) data required for implementation and maintenance of the SSES EQ program. EQDB change requests are processed to support plant modifications, Replacement Item Evaluations (RIEs), maintenance, procurement activities, and for resolution of issues. The EQDB implementation, use, and maintenance is performed in accordance with procedural guidance. Changes to EQDB data are made from either a new or revised Environmental Qualification Assessment Report (EQAR), or by an EQDB Change Notice. EQARs are controlled design output documents which contain information adequate to establish the environmental qualification of SSES components. Updates are performed in accordance with procedural requirements.

The Dynamic Qualification of Equipment/Seismic Qualification Review Team (SQRT) program assures that common cause and/or common mode failures of equipment do not result in simultaneous failure of redundant safety systems due to a seismic and/or hydrodynamic event. The modification program can result in new SQRT Binders

being developed or SQRT Binder Change Notices being generated against existing SQRT Binders. This process is performed in accordance with procedural requirements. Changes to SQRT binders are accomplished using a SQRT Binder Change Notice (SBCN). Changes may be issued independently or with a modification package, as appropriate.

Installation, Operating and Maintenance Manuals (IOMs), which describe structures, systems, subsystems, components and subcomponents of SSES, are required to be updated due to the modification process in accordance with procedural requirements. Changes to IOMs are made using the IOM Change Notice (IMCN). Changes associated with a plant modification are created, reviewed and approved as part of the modification package, and posted when the modification is installed. Changes which are not associated with a modification are approved by the appropriate system engineer and posted.

2. Plant Modification Processes

Nuclear Department policy with regard to programs and projects which modify the plant design and configuration, except for core design changes, is defined in the current Nuclear Department Modification Program. The program has undergone evolution since PP&L accepted design responsibility. Even though the program has been in compliance with the regulations throughout operation, PP&L has made organizational and process improvements to strengthen the program. The modification program applies design and configuration controls to plant design and configuration changes, as defined below.

- PP&L defines a "design change" to be a change to the plant configuration that requires or results one or more design inputs or which requires formal (i.e., procedurally documented, reviewed and approved) calculation and/or analysis to establish that the original design inputs are still met.

- PP&L defines a modification as a planned change to plant configuration or design output documents, and accomplished in accordance with the requirements and limitations of applicable

codes, standards, specifications, licenses and predetermined safety restrictions.

PP&L procedures provide guidance for determining when a design change is occurring in addition to a configuration change. Controls for reactor core design changes are delineated in specific Nuclear Fuels procedures.

PP&L utilizes a plant modification process, in which there are variations in the implementation of five types of modifications. The following discussion has been organized to address the description of each process, including the implementation of 10 CFR 50.59, 10 CFR 50.71(e), and 10 CFR 50 Appendix B. This illustrates use of the processes, as well as compliance to the regulations.

At PP&L, changes to plant hardware design and configuration are made using one of five types of modifications: the Design Change Package (DCP), the Engineering Change Order (ECO), the Setpoint Change Package (SCP), the Replacement Item Evaluation (RIE), and the Bypass. Each is an adaptation of the generic process described above, and is subject to 10 CFR 50 Appendix B, 10 CFR 50.59, and 10 CFR 50.71(e) requirements.

10 CFR 50 Appendix B requirements are integrated into the modification procedures. The procedures for each of the types of modifications invoke the requirements of 10 CFR 50, Appendix B, Criterion III. Since each type varies in the potential effect on the design bases, design output documentation, and design complexity, the process allows a graded approach to implement the conservative procedural requirements that are invoked for Design Change Packages (DCPs).

The following discussion is intended to give a process description which illustrates the identification and control of design interfaces required by 10 CFR 50 Appendix B, Criterion III.

a) *Design Change Package (DCP)*

A DCP is a specific documentation package required for modifications which require 10 CFR 50.59 Safety Evaluations, are complex, or result in a change to the plant design. The DCP lists coordinated design output documents required to implement a design change to plant systems, structures, and components. The requirements for preparation of DCPs are described procedurally. The procedure identifies responsibilities and activities required for preparation, review, approval, issuance and revision of DCPs.

The identification of the regulatory requirements and design bases is provided through the use of design inputs. The development of the design inputs is proceduralized. The DCP utilizes the complete "Design Input and Design Consideration Checklist" with appropriate reference to the associated Design Guide. The design is supported by calculations. The selection of materials is performed and controlled. The process identifies and controls the design interfaces. Design verification is performed for design output documentation and calculations.

DCP installation and close-out activities are also procedurally defined. This process, which is integrated with other applicable procedures, includes: providing input to the plant schedule; preparing the Plant Modification Package; developing the work authorization; ordering material; monitoring the installation; resolving installation problems; overseeing functional testing; closing out work authorizations; and initiating close-out of the DCP and record requirements. Operating, maintenance, and testing procedures are revised and updated through the modification program. Licensing documents, including the FSAR, are also required to be reviewed for impacts from the proposed change and updated in accordance with governing procedures.

b) *Engineering Change Order (ECO)*

An ECO is a modification used to implement relatively simple configuration changes to plant systems, structures,

and components that are not design changes. The ECO process may not be used to install modifications under the following conditions: (1) when the change alters the manner in which a system operates; (2) when a 10 CFR 50.59 Safety Evaluation (see Section II.C2.b) is required; or (3) when significant engineering man-hours are required.

The requirements for preparation of ECOs are described in the same procedure as for DCPs. The procedure identifies responsibilities and activities required for preparation, review, approval, issuance and revision of ECOs. Given the limitations on ECO use described above, changes to licensing documents are editorial or administrative in nature.

c) *Setpoint Change Package (SCP)*

An SCP is a modification to implement a setpoint change, which is a specific type of design change that alters preset values of instrumentation or devices which perform, prevent, or initiate predetermined actions. Setpoint changes may also be implemented as part. of a DCP. The requirements for the preparation of SCPs are described in a written procedure. This procedure establishes the program for selection, control, and documentation of permanent setpoints for bistable and adjustable devices used in instrumentation, process systems, control, alarm, power, transformer taps, protective circuits, and motor operated valves at SSES. Setpoint changes do not generally require equipment or material procurement; or removal, addition, or replacement of field equipment. SCPs are required to be processed in accordance with 10 CFR 50.59. Licensing documents, including the FSAR, are also required to be reviewed for impacts from the proposed change and updated in accordance with governing procedures. The framework for design control for a setpoint change is analogous to that used for a DCP, as described above.

The identification of the regulatory requirements and design bases is provided through the use of design inputs. The development of the design inputs is proceduralized. The SCP utilizes the "Design Input and Design Consideration Checklist" with appropriate reference to the

associated Design Guide. The design is supported by calculations. The process identifies and controls the design interfaces. Design verification is performed for design output documentation and calculations.

d) Replacement Item Evaluation (RIE)

An RIE is a modification which may be used when an exact replacement for an existing component is not available or is not desirable. The RIE process may not be used to install replacement items under the following conditions:

- to add or delete plant components or subcomponents;
- when a 10 CFR 50.59 Safety Evaluation (see Section II.C2.b) is required; or
- when the replacement constitutes a plant design change.

The requirements for preparing RIEs are described in a written procedure. The procedure defines the process for evaluating non-identical replacement items for use at SSES, including a comparison of critical characteristics, the identification of installation requirements, limitations, and maintenance of configuration control to assure maintenance of the design bases.

The process identifies the "critical characteristics" for design, which are those properties or attributes essential to the item's form, fit, and functional performance. Critical characteristics for design are the identifiable and/or measurable attributes of a replacement item, which provide assurance that the replacement item will perform its design function. This effort provides for the identification of the regulatory requirements and component design bases. The process does not allow an RIE to make a plant design change, therefore calculations are not normally required. The process identifies and controls the design interfaces required for this limited effort. The applicable design output documentation as identified receives independent verification.

This RIE process requires the engineer to identify impacts to design output documentation, (e.g., drawings, specifications, etc.), and to implement appropriate changes

in accordance with the governing procedures. Given the limitations on RIE use discussed above, changes to licensing documents are editorial or administrative in nature.

e) *Bypass*

A bypass is a temporary change to a mechanical component, electrical component or instrument in a power plant system that is planned to be restored to original configuration. A bypass is not a substitute for a permanent design change. Bypasses are required to be processed in accordance with 10 CFR 50.59. Typical bypasses include jumper wires, open states links, lifted leads, removed fuses, installed hoses, blind flanges, spool pieces, temporary power or instrument/electrical temporary setpoint changes. The bypass mechanism is used to:

- obtain appropriate engineering review of maintenance troubleshooting activities;
- alleviate abnormal equipment operation;
- solve an existing problem which may jeopardize safe or continuous operation;
- provide a method to temporarily adjust plant equipment in order to compensate for various operating conditions;
- isolate a defective component from the plant until the defective component is replaced; and/or
- test new configurations of systems/components prior to permanent design changes.

Preparation requirements and constraints for bypasses are defined procedurally. This procedure, in conjunction with the use of qualified and experienced personnel, training, and the QA/QC oversight assures: preservation of plant safety, reliability, and configuration control; operator awareness; and conformance with design intent and operability requirements.

The identification of the regulatory requirements and design bases is provided through the use of design inputs. The development of the design inputs is proceduralized. The Bypass process utilizes a list of design considerations which is included in the bypass procedure. The process

identifies and controls the design interfaces. The applicable drawings identified by the procedure have a change mechanism added to reflect the change which will remain until the bypass is removed. The bypass and drawing changes are reviewed by a second qualified engineer.

Nuclear Systems Engineering conducts a semi-annual assessment of bypasses open for more than six months. This assessment is provided to the Plant Operations Review Committee for their review.

Bypasses typically do not require changes to licensing documents, including the FSAR, due to their temporary nature. If an installed bypass is to be made permanent, the proper modification process must be followed, including licensing document updates.

3. Reactor Core Reload Design Process

PP&L's role in the area of reload design and licensing analyses has been expanded throughout the plant construction, start-up testing, power ascension, and commercial operation phases. This was warranted by the fact that the first reload of each unit marked a transition to a new fuel vendor. This change required PP&L to understand the data requirements and analysis capabilities of both fuel vendors in order to determine and assimilate the necessary data to support the new vendor.

Beginning with the second cycle of each unit, PP&L assumed the responsibility for preparing cycle-specific license amendment requests for submittal to the NRC for review and approval. PP&L was active in the review and approval of all cycle-specific fuel vendor documents that were necessary to support license amendments. This process of preparing and submitting cycle-specific license amendments further broadened PP&L's in-house knowledge and expertise in the reload design and licensing analysis areas.

During this time, PP&L undertook a significant effort to develop, validate, and benchmark the in-house three-dimensional simulation and plant system models necessary to perform in-house reload design and licensing analyses. The development and qualification of the in-house expertise was an integral part of the overall effort to

obtain NRC approval. In 1992, PP&L was approved by the NRC to perform in-house reload design and licensing analyses. In 1993, the NRC approved the use of a Core Operating Limits Report to support reload cycle specific changes for each unit.

FSAR, Section 17.2, states PP&L's commitment to comply with the Quality Assurance requirements of 10 CFR 50, Appendix B. This commitment is reiterated in Nuclear Department Operational Policy Statements, and is incorporated into PP&L's reactor core reload design process by reference. Nuclear Department Operational Policy Statements identify the elements of the design control process for nuclear fuel design and licensing. Nuclear Department procedures identify the organizational responsibilities that have been defined within Nuclear Fuels to implement the requirements of the Operational Policy Statements.

The reload core design process is controlled via Functional Unit Procedures. These address the items listed below:

- The Reload Design and Analysis Program controls PP&L's reload design and analysis process to ensure adherence to applicable regulatory and design requirements. This procedure describes the detailed scope of the reload design and licensing processes, defines design responsibilities, and identifies the documentation, review, and approval requirements for reload design and licensing analysis activities.

- The Documentation of Analyses procedure defines the format, content, responsibilities, and personnel qualifications necessary for documentation and independent verification of quality-related analyses.

- The Nuclear Fuels' procedure on software products defines the software quality assurance requirements for software products utilized in a quality-related analysis. These requirements include documentation, review, approval, and modification control.

- The procedure for preparation, review, and approval of a Core Operating Limits Report (COLR), defines the procedural controls for a COLR, including review and approval.

The overall responsibility for the reactor core reload design process resides within Nuclear Fuels. Each member of the Nuclear Fuels Engineering staff is individually qualified to perform reload design

and licensing analysis related activities. This qualification program supplements the Nuclear Department Engineering Support Personnel Training Program previously discussed in this section.

The Reload Design and Analysis Program requires development of a specific reload design envelope, for each reload design, which defines the design inputs (e.g., licensing limits, mechanical design criteria, operational considerations) that are used to validate the reload core design. The reload design envelope is contained in the reload-specific Reload Design Plan which is approved by the Supervisor - Nuclear Fuels Engineering. The proposed cycle-specific reload design is reviewed by the Reload Design Review Board (RDRB), and if found acceptable, recommended for approval by the Manager-Nuclear Fuels. The Reload Design Review Board is a subcommittee of the Engineering Review Committee (see Section V) and is an advisory committee for the Manager - Nuclear Fuels and the Manager - Nuclear Engineering. The RDRB is composed of senior members of the Nuclear Fuels staff and a member of the onsite Reactor Engineering group. RDRB review and Manager - Nuclear Fuels approval is required prior to formal release of a reload design to the fuel vendor for review and concurrence.

Similar to the Reload Design Plan, a Reload Analysis Plan is developed to govern the activities necessary to license a reload design. The Reload Analysis Plan includes documentation of a review of the Licensing Basis Analyses to identify those required to be analyzed for the reload design of interest. The Reload Analysis Plan also includes documentation of a review of all plant modifications which have required Nuclear Fuels review. The Plant Modification Process and the Applicability Criteria for Design Considerations Design Guide trigger review by the Nuclear Fuels group for proposed modifications which could affect the analyses performed as part of the reactor core reload design process.

Another key aspect of the reactor core reload design process is oversight of the fuel vendor. PP&L maintains responsibility for the overall process and, as such, is active in the review of vendor-generated documents which support each reload core design. The scope of these reviews is discussed in the Reload Design and Analysis Program.

The reactor core reload design process integrates reload licensing analysis activities which are performed within Nuclear Fuels with the requirements of other Nuclear Department Administrative Procedures (NDAPs). The Reactor Core Reload Design process requires the preparation of a 10 CFR 50.59 Safety Evaluation to support implementation of all reload core design changes. These Safety Evaluations are prepared, reviewed, and approved in accordance with the NDAP. If a license amendment is required to support implementation of a reload core design change, the Reactor Core Reload Design process references the NDAP on Implementation and Control of License Amendments for preparation, review and approval of proposed license amendments. The Reactor Core Reload Design Process requires the preparation of changes to the FSAR for incorporation of the appropriate information from the reload cycle specific documentation. These FSAR changes are prepared in accordance with the NDAP for controlling changes to licensing documents.

The cycle-specific reload licensing analyses are similar to the reload design analysis reviewed by the RDRB. Following an approval by the Manager-Nuclear Fuels, a set of reload cycle-specific documentation (e.g., Reload Summary Report, Safety Evaluation, cycle specific Core Loading Map) is prepared by Nuclear Fuels and reviewed by appropriate organizations. This documentation, containing the reload cycle-specific Safety Evaluation, is submitted to PORC in support of PORC review and approval of the reload cycle-specific Safety Evaluation. SRC review and approval is obtained as required.

E. Description of Configuration Control Processes

Configuration control processes meet the requirements of 10 CFR 50, Appendix B, Criteria III, V, and VI, and maintain consistency between the representation of the plant in design output documents and the physical plant. They also provide mechanisms to identify, resolve, and correct inconsistencies.

Plant modifications have the greatest potential impact on design output documents and the physical plant. Configuration control of the operating plant is maintained by experienced, knowledgeable, and qualified personnel with procedural control which includes status control, operator check-off lists, surveillance testing, alarm response, normal and off-normal operating procedures, and emergency procedures. Configuration

control of the reactor core is governed by the reactor reload core design process and related procedures and is discussed below. The final process in configuration control involves the maintenance of the design output documents.

1. Configuration Management of the Plant

PP&L has established a configuration management program that is responsive to the Operational QA Program. PP&L's configuration management program is defined in procedures and includes the items listed below.

- The facilities of SSES to which the configuration management program applies include: Unit 1, Unit 2 and common systems, including the Radwaste System, Diesel Generators, the simulator, spare parts stock, the Low Level Radwaste Holding Facility, North and South Security Gatehouses, the Warehouse Tie Structure, the Security Control Center and the Engineered Safeguards Service Water Pumphouse.

- The managed information sources are sources of plant configuration information which are developed, maintained, and controlled by qualified personnel using approved procedures that must be maintained to accurately represent the as-built plant. A listing of SSES managed information sources is delineated in a Nuclear Department procedure.

- The requirements and responsibilities for maintaining these managed information sources are delineated in a Nuclear Department procedure. The configuration management program assures accurate communication about the design, design bases and design parameters to those who implement the plant design through operations, maintenance, procurement and training activities.

The program assigns responsibility for each managed information source to an individual group, which establishes the controls for generation, approval and maintenance of the information, appropriate to the type of information.

Controlled information sources are maintained in either an as-engineered condition or in an as-built condition. Information sources maintained in an as-engineered condition are provided with

a change mechanism that assures a level of review and approval suitable to the specific information source, and establishes the information update requirements so that the updated information is available upon approval. Information sources maintained in an as-built condition are controlled such that the as-engineered condition is shown and tracked, and the as-built condition is shown only after the actual change has been made. Department processes which affect managed information sources incorporate the established mechanisms into the processes to reflect information changes.

All managed information sources are provided with the means to identify errors to the information and update the information sources to reflect correct information. Changes to managed information sources are done in a manner which is appropriate for the significance of the change being made.

QA audits are performed of the configuration management program and the various processes that impact the configuration management. A review of past audits shows that the program and processes have evolved based upon feedback from oversight of process implementation. When problems were identified, they were resolved via the PP&L corrective action program.

2. Configuration Management in the Plant Modification Process

Plant configuration is altered via one of the methods discussed in Section II.D. That section defines the plant modification process and references the controlling procedures. Each of the five modification types have procedurally-defined processes which contain configuration control requirements suitable to the specific modification process. The following major configuration control requirements are applied to the plant modification process:

- Individual modifications must be uniquely identified, packaged and their scope clearly identified.

- The configuration information impacted by the modification must be identified.

- The change mechanism appropriate to each of the impacted, managed information sources must be prepared, reviewed and approved, and included in the uniquely identified package.

- The completed modification package is transmitted to Nuclear Records for logging into the configuration control database and statusing of the associated change mechanisms.

- Changes to the modification after release of the package to Nuclear Records must be identified and tracked against the managed information sources.

- When the modification is completed, the configuration control database is updated to reflect completion and the change mechanisms, including changes made during the installation process, are distributed so the affected, managed information sources reflect the as-built condition.

- After the modification is complete, the configuration management program directs each group responsible for a managed information source to update the managed information source in accordance with the procedure(s) governing that information source.

These steps are an integral part of the modification process. Thus, at the end of any modification cycle, the modification process itself provides reasonable assurance that the managed information describing the plant matches the configuration of the physical plant itself.

QA audits and surveillance, as well as assessments of the Modification Program, have determined that process controls conform to QA Program requirements. Over the years, the modification process has evolved. Cited problems with the modification program were, and continue to be, resolved via the corrective action program.

3. Configuration Control of the Operating Plant

Nuclear Department procedures describe the duties and responsibilities of SSES operations personnel, including the responsibilities and requirements for preparation, review, approval and control of the operations procedures. These procedures provide the required configuration of the plant equipment for its different modes of operation. The procedures are developed and are responsive to design basis changes to the plant. Refer to

Section III for a description of the control of operating, maintenance and testing procedures.

The routine configuration of the plant equipment is controlled by the various operating procedures and the Check-Off Lists (CLs) which references appropriate design output documentation. In order to start a system initially, a certain configuration is required. The CLs show this 'normal' system configuration that is established prior to starting the system. From that point on, the various system operating procedures define the system configuration and changes to the system configuration applicable to specific operating conditions. Thus, these two documents establish the routine operating plant configuration.

For example, the Reactor Heat Removal (RHR) system operating procedure references a number of system CLs that must be satisfied in order to ready the RHR System for operation. The procedure describes system reactions to automatic operation and required operator actions for manual operation. Thus, the procedure provides the ongoing configuration control requirements for different modes of operation.

Once a system is operating, equipment may need to be removed from service to perform testing, corrective maintenance, preventative maintenance, and modifications. Situations may occur in which systems are put into conditions not covered by the operations procedures. These alignments are evaluated and tracked by a process defined within the Nuclear Department procedure for system status and equipment control. This process documents the status of system components that are not in agreement with the CLs or OPs. The procedure for system/equipment release provides the mechanism to evaluate the anticipated configuration of the plant during the period in which equipment is released for maintenance, in order to assure the plant still meets the design intent and is safe.

The surveillance testing program establishes the administrative controls for implementation and maintenance of surveillance procedures and tests used to satisfy the requirements identified in the SSES Technical Specifications. Technical Specifications, as Appendix A to the Operating License, set forth the limits, operating conditions, and other requirements imposed upon facility operation for the protection of the health and safety of the public. They are derived from the analyses and evaluations included in the

SAR, and amendments thereto. Many of the system functional design bases are included in the Technical Specifications.

Procedures require that testing preserve the validity of the Initial Test Program, and ensuing plant modification testing. The testing program considers all requirements of the current Technical Specifications, FSAR and the plant operating license. These procedures also requires the determination of functional testing, impact on existing operability tests, and performance testing.

Nuclear Department procedures also describe the duties and responsibilities of maintenance personnel. The maintenance program is accomplished by performing predictive, preventive, planned and corrective maintenance in accordance with the proper balance of scope-setting, priority, planning and scheduling. The Work Authorization (WA) procedure provides a system to ensure that work activities associated with plant systems, structures, and components are identified, controlled, and documented. The WA system is used to control corrective and preventive maintenance, and to implement approved plant modifications (e.g., DCPs, ECOs, SCPs, RIEs, and bypasses).

Plant Modifications have the greatest potential impact on the design bases, operating, maintenance and testing procedures and the managed information sources of SSES. Therefore, QA practices and policies have been established to assure that as the plant is modified, the plant operating, maintenance, and testing procedures are reviewed by knowledgeable staff and engineering personnel and are appropriately updated to translate the design change. The operation, maintenance and testing procedures are discussed in detail in Section III.

QA audits and surveillances have been conducted of plant operations, material control, status control, and Technical Specification compliance programs, and have determined that the processes conform to the QA program. Independent and self-assessments have also been conducted in these areas and have reached similar conclusions.

4. Configuration Management of the Reactor Core

The configuration of the reactor core is controlled through a combination of engineering and operations (reactor engineering) FUPs. The Reload Design and Analysis Program establishes the controls for preparation, review, and approval of the reload core cycle specific Core Loading Map. This Core Loading Map is utilized in establishing the change process for the reload core per the FACCTAS Preparation Guidelines. Guidelines for performing the physical work related to the reload core design change are presented in the refueling operations procedure. Finally, verification of the reload core is performed using the Core Loading Map (generated under the Reload Design and Analysis Program) and the Core/Fuel Pool Verification procedure. This process ensures that the final reactor core configuration is consistent with the analyzed reactor core design.

Once again, the QA audits and surveillances of fueling and refueling activities have determined that processes conform to the QA program.

It should be noted that PP&L and NRC did find problems associated with overall work control of refueling floor activities and the control and operations of the refueling bridge. However, none of these problems indicated a negative impact on core configuration. Via the corrective action program, PP&L evaluated the significance of the problems and implemented corrective actions through the establishment of a dedicated responsible manager for all refuel floor activities, retraining of the operations staff assigned refuel bridge operations, and the implementation of an upgraded refueling platform and control system.

F. Conclusion

The original design, construction, procurement, licensing, and start-up activities were conducted under quality assurance controls and practices that implemented 10 CFR 50, Appendix B. These activities established a solid foundation for the current control system. Personnel gained knowledge of the SSES design bases through active participation with the design.

Since that time, PP&L has maintained and expanded design, configuration management, and document control processes to assure integrated processes for managing changes to the original design baseline. Our current integrated set of quality assurance principles, processes and procedures used to control changes to, the design and configuration of the plant are responsive to governing requirements and provide mechanisms for translating design bases changes into the plant configuration. These processes have evolved over time and reflect experiences gained throughout our own operating history, as well as from industry events.

Application of quality assurance principles coupled with the solid original design baseline provides reasonable assurance that PP&L is appropriately managing the design and configuration of SSES.

III. Response to Item (b): Translation of Design Into Operating, Maintenance and Testing Procedures

A. Introduction

1. Purpose

This section of the response provides PP&L's:

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

2. Overview

Procedure development and revision can be considered in a manner similar to the discussion in Section II regarding the establishment of the original "baseline" design for the Susquehanna Steam Electric Station (SSES) and the "change management" processes that preserve the design. The initial procedure development captured the baseline design and the change management processes preserve the design. This section provides a summary description of the translation of the original plant design into plant procedures, existing processes for continuing the translation of plant modifications, and verification activities that confirm the proper translation.

PP&L's controls for operation, maintenance, and test procedures were established to meet the requirements of 10 CFR 50, Appendix B, specifically, Criteria V, VI and XI. Specific portions of the PP&L Quality Assurance (QA) Program establish the policies and processes for the preparation, review, approval, distribution, use, and change of Nuclear Department operation, maintenance and test procedures. These requirements were established in the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR), for initial start-up activities and operational activities, respectively. Requirements were further delineated within the PP&L QA Manual in place during initial

design, construction, and testing activities, and are currently delineated within Operational Policy Statements and associated Nuclear Department Administrative Procedures (NDAPs). The same procedural requirements and processes are generally used for procedures that are not implemented under the Operational QA Program.

3. Organization

The detailed response to item (b) addresses the following topics:

- Translation of Original Plant Design Into Plant Procedures
- Mechanisms for Assuring Proper Translation of Design Changes
- Verification Activities Which Confirm Proper Translation

B. Translation of Original Plant Design Into Plant Procedures

As described in Section II, the Initial Test Program, controlled under the scope of the PP&L QA Program, commenced with structure/system/component turnover, and terminated with the completion of power ascension testing. The translation of original plant design into plant procedures was accomplished by operating, maintenance and instrumentation personnel utilizing the same design bases information used to support the Initial Test Program. These concurrent activities provided an opportunity to exchange information, such that the original plant design translated into the Initial Test Program was also translated into plant procedures.

PP&L plant maintenance, instrumentation, and operations personnel were utilized in the performance of the component inspection and testing during the Initial Test Program. This "hands-on" experience assured an understanding of the plant design on the component level. This understanding was directly translated into plant maintenance and testing procedures that were being prepared in a parallel effort to the Initial Test Program.

Plant Staff/Engineering personnel were involved with the preoperational/acceptance testing. In accordance with regulatory requirements, test procedures demonstrated, to the extent practicable, the capability of safety and non-safety related structures, systems and components to meet their performance requirements. The tests included initial status of support systems, and detailed alignments for system

valves, circuit breakers, switches and instruments. Typically, plant engineers reviewed the preoperational/acceptance test procedures. In many cases, the procedures provided direct input for the preparation of the system operating procedures. This involvement by PP&L personnel with the appropriate PP&L, Bechtel and General Electric (GE) personnel assigned to the Initial Test Program, helped to provide assurance that the initial operating, maintenance, and testing procedures accurately reflected the design bases.

Start-up testing was implemented by plant and engineering personnel. Start-up tests confirmed the design bases for operational and functional attributes that can be validated by test. In addition, the program demonstrated, to the extent practicable, that the plant would operate in accordance with design, and was capable of responding, as designed, to anticipated transients and postulated accidents.

The adequacy of plant operating and emergency procedures was confirmed by use during start-up testing. These tests included control system tune-ups, nuclear instrumentation surveillances, and system surveillances which demonstrated proper plant response to self-induced minor transients, such as turbine valve testing. Tests which collected and analyzed data to demonstrate proper steady-state system performance were conducted after the systems were aligned, using permanent plant procedures. "Once-and-done" testing, such as that performed for major transients or piping thermal growth verifications, was conducted using start-up tests expressly written for those demonstrations.

PP&L engineering and operations personnel, along with vendor representatives, reviewed and approved procedures and results as an integral part of the Test Review Committee, a subcommittee of the Plant Operations Review Committee (PORC).

Prior to performing tests which could have potentially resulted in a plant transient, on-shift operations personnel were required to review applicable emergency procedures. Plant operating and emergency procedures also received an additional documented review from GE, the Nuclear Steam Supply System (NSSS) supplier. This specific review was required by the NRC as part of our initial licensing process.

As described in Section II, the engineering turnover process provided specific design bases information on a system-by-system basis. This data was utilized by engineering, plant staff, and nuclear training personnel during the procedural development and validation processes, as well as for the development of specific training modules. This provided additional

assurance of the translation of the design bases information into the operating, maintenance, and testing procedures.

The design bases for the reactor core implement 10 CFR 50, Appendix A requirements, with compliance demonstrated by GE for the initial reactor cores for both SSES Units. The translation of these original design bases for the reactor core into operational requirements was accomplished through the establishment of Technical Specification limits, surveillance requirements, and surveillance frequencies to maintain system operability. These Technical Specifications were translated into maintenance, operation, and testing procedures as part of the process discussed above.

C. *Mechanisms for Assuring Proper Translation of Design Changes*

With this process baseline established, the issue becomes one of maintaining the procedures to accurately reflect any changes that occur through modifications to the physical plant or changes in operating and maintenance philosophy. PP&L, through the establishment of quality assurance policies and practices, has implemented processes that will assure the adequacy of this baseline. A key ingredient in the established processes is the multi-disciplined reviews of procedures, including those by one or more of the Nuclear Department advisory committees prior to approval by management for distribution for use. Mechanisms for assuring proper translation of design changes are discussed in the following subsections.

Plant modifications have the greatest potential impact on operating, maintenance and testing procedures, managed information sources and the design bases of SSES. Therefore, processes have been established, consistent with quality assurance principles, to assure that as the plant is modified, procedures are reviewed by knowledgeable personnel and are appropriately updated to translate the design change.

These processes are subject to periodic review as a part of the formal periodic audit program established in response to 10 CFR 50, Appendix B, Criterion XVIII, to evaluate adequacy and effectiveness. Early in the life of certain of these processes, there were administrative problems which indicated some weaknesses. However, none of the weaknesses indicated broad failure to translate the design bases into the appropriate plant procedure. Rather, the problems tended to be associated with the timeliness of some updating of procedures following implementation of modifications. These problems were adequately addressed via the PP&L corrective action program (see Section V).

1. Control of Procedures

PP&L has had and continues to have processes in place to assure that operating, maintenance, and testing procedures are properly controlled, including accurate reflection of the design bases. The following subsections provide a discussion of the general controls for operating, maintenance, and testing procedures. Via NDAPs and Functional Unit Procedures (FUPs), PP&L has defined the processes for preparing, reviewing, and approving procedures (and changes thereto) required for the conduct of activities associated with the operation, maintenance and testing of SSES.

Procedure content and format comply with applicable portions of Regulatory Guide 1.33, Revision 2, SSES Technical Specifications, and PP&L's commitments to industry guidance (e.g., ANSI N18.7-1976). Procedure development has been enhanced by the first-hand knowledge gained by PP&L through the various turnover and testing activities discussed above. Since initial development of the procedures, the requisite quality assurance practices include periodic reviews to determine if changes are necessary or desirable. Procedures subject to this review are delineated in Nuclear Department procedures.

In addition, procedures are reviewed when significant system or equipment modifications are made, and following an unusual event, such as an unexpected transient, a significant operator error, or equipment malfunction where the procedure contributed to the cause of the event, or was inadequate in mitigating the effects of the incident. A Nuclear Department procedure defines the review of plant procedures, and intent changes thereto, by the PORC and approval by the Plant Manager-SSES.

Changes to operating and maintenance procedures are evaluated in accordance with 10 CFR 50.59 (see Section II). The effect of procedure changes on design bases and configuration control is also addressed through the involvement of knowledgeable, personnel trained in the procedure preparation, review and approval processes.

In order to ensure the integrity of the plant, it is necessary that testing be complete such that the Initial Test Program and all ensuing plant modification testing remain valid. Thus, department procedures require that testing include consideration of the current Technical Specifications, FSAR and the plant operating license

(including all amendments). Procedures also require the identification of necessary functional testing, and any impacts of proposed plant modifications on the existing operability and performance testing. FUPs are in place to assist the System Engineer in preparing and planning tests. Emphasis is placed on nuclear safety, testing design intents, assuring configuration control, and preserving operational performance. Post-modification testing requirements are established by the modification team (see discussion below), and reflect the "as-designed" modification.

2. Implementation of Plant Modifications

PP&L has established plant modification and reactor core design change processes. How these processes assure updating of the plant operating, maintenance, and testing procedures is discussed in the following subsections.

The Nuclear Department modification program, described in Section II above, has established quality assurance practices that implement 10CFR50, Appendix B which assure that changes to the plant design and/or configuration are properly incorporated into operating, maintenance and testing procedures. The following discussion details how these practices assure that necessary procedure changes are identified and incorporated during the implementation phase of the five types of plant modifications, which include: the Design Change Package (DCP), the Engineering Change Order (ECO), the Setpoint Change Package (SCP), the Replacement Item Evaluation (RIE), and the Bypass.

DCP preparation and installation requirements are delineated in department procedures. As the modification design proceeds to completion, installation, testing and closure strategies are developed. The detailed work to be performed is also established, including component functional testing, system operability and/or applicable performance testing, and procedures requiring change prior to system operability. This information is documented in the Plant Modification Package which controls the implementation, testing and site closure of DCPs.

Following installation, modification closure includes a final verification that procedures required for operability have been changed and issued. ECOs follow a similar implementation process as described above for the DCPs, although due to the nature of an ECO, very few procedural changes are needed.

A "Setpoint Change Implementation Package" (SCIP) is required by department procedures to be prepared by the System Engineer upon issuance of an SCP to Nuclear Records. The SCIP includes a "Procedure Change Sheet," which lists all procedure changes required as a result of the setpoint change. Upon implementation of the setpoint change, all procedure changes are processed by the work group installing the change or the System Engineer, as appropriate.

In accordance with a Nuclear Department procedure, the RIE process is only used to replace existing components or subcomponents, (i.e., cannot add or remove components or subcomponents) cannot require a 10 CFR 50.59 safety evaluation and cannot permit a design change. Because of the limited nature of an RIE, when implemented, it does not change equipment device numbers nor the component function, and procedure changes are typically not required. On occasion an RIE may conclude that the non-identical replacement item is adequate, but the installation requires system or supporting changes that extend beyond the immediate interface with the replacement item. When this situation occurs, the RIE may be used to procure the item, but either a DCP or an ECO would be required for installation.

Following the development of a bypass package (see Section II), a "Bypass Installation Form" is developed to identify the required procedures which require revision. Once installation has been completed, the System Engineer confirms that appropriate testing has been completed and that procedure changes are issued. When it is determined that the bypass should be removed, the installation process is essentially reversed by preparation of a "Bypass Removal Package," which includes the "Bypass Removal Form." This package assures that all required testing and procedure changes are prepared by the System Engineer to reflect the removal of the bypass.

3. Implementation of Reactor Core Design Changes

Implementation of reactor core design changes is accomplished procedurally. The reload design and analysis program requires the issuance of a cycle-specific reload core loading pattern. This core loading pattern translates the design into a document which is a prerequisite for SSES plant procedures governing the cycle-specific configuration of the reactor core. (The SSES plant procedures governing core configuration were previously discussed in Section II). Change control, including translation of design requirements into the applicable SSES plant procedures, is discussed below.

Additional design requirements resulting from the reactor core design change process are represented in the reload cycle-specific "Core Operating Limits Report (COLR)." The reload cycle-specific COLR is a term defined in Technical Specifications and is generically referenced throughout. Incorporation of the COLR into operating, maintenance, and testing procedures is accomplished by reference to the applicable SSES Technical Specifications. The preparation, review, and approval of a COLR is procedurally controlled.

D. Verification Activities Which Confirm Proper Translation

PP&L has a number of on-going verification activities and has had major projects which provide assurance of the proper translation of the design into operating, maintenance and testing procedures. The examples discussed in the following subsections include audits and assessments, Power Uprate Program, DBD Project, ITS Project and a major program to upgrade the SSES Emergency Operating Procedures. The specific activities referenced are only a sampling of the activities actually performed.

1. Audits and Assessments

A level of assurance is provided through periodic audits and surveillances of the processes, as well as the in-line quality assurance document reviews of new or revised plant procedures. During audits the adequacy of the governing procedures and programs is evaluated against licensing commitments and technical content, including design bases. Historically, the content of procedures subject to the audit and surveillance process has been

found to be consistent with commitments and technical bases. As with any program or process under the scope of the QA program, when a problem is identified it is handled within the requirements of the corrective action program.

This QA coverage has continued over our operating history. Examples illustrate the types of reviews involving the translation of design into procedures which have been conducted. (Note that these are listed only to illustrate the types of activities performed; numerous audits have been performed in these areas.) A 1982 audit evaluated five safety-related modifications. One of the conclusions was that the design packages properly identified procedures to be revised. A 1984 audit specifically looked at the proper updating of operating procedures effected by modifications. The audit revealed that for the cases examined, procedures were being updated to identify changes made to the plant. A 1992 audit of the setpoint change program found that in the case of seven setpoint change packages, the applicable procedure change sheet was completed, as required by the governing procedure.

Another level of assurance is provided through assessment activities. Many of these activities are coordinated through the Independent Safety Evaluation Services Group, the Nuclear Assessment Services Group and the Susquehanna Review Committee (SRC). These assessments have served as opportunities to identify and act upon opportunities for improvement, including assuring that specified quality and safety requirements are met. A sampling of specific assessments involving design bases, testing and the translation of design into procedures are listed below. (Note that these are listed only to illustrate the types of activities performed over the years; additional assessments have been performed in these areas.)

- An independent assessment of plant design, testing and procedures was performed in 1980. This was a comprehensive assessment of major safety-related systems. The majority of the recommendations were for improvements in operation and maintenance, rather than for corrections of safety or design deficiencies.

- An assessment of the readiness to load fuel and safely operate the unit was performed prior to initial fuel loading on each unit (1982 and 1984). These assessments included a specific

observation that the preoperational testing was competently and correctly performed.

- An assessment was conducted in 1982 during the preoperational testing of three safety-related systems. It was concluded that the tests were conducted in a competent manner and acceptance criteria were met.

- A 1992 assessment was performed concerning the use of Alternate Shutdown Cooling. This assessment started with the licensing bases established in the FSAR, and reviewed the adequacy of available equipment, procedures and training. Recommendations were made to enhance the procedure and training. The recommendations were accepted and implemented.

2. Power Uprate Program

The Power Uprate Program (PUP) was initiated in 1991 to increase the power output of both SSES units by approximately 5%. The objective was to apply the design margin that existed between the originally licensed thermal power and the power level for which the plant equipment had been designed and sized. Completion of the PUP required amending the operating license and implementing numerous changes to the plant configuration, design basis information and plant procedures. In 1994, SSES Unit 2 achieved a successful power uprate of about 5%. Unit 1 was uprated in 1995.

A critical part of the PUP was the review conducted to ensure the capability of plant systems and structures affected by the uprated conditions. Section VI describes the extensive nature of this review. A separate calculation was established for each system, structure or topical review. Each review document contained an assessment of the system's or the structure's ability to perform its design function at uprated conditions. The system, structure and topical reviews conducted by PP&L and General Electric formed the technical basis for the SSES Power Uprate Licensing Topical Report. Following the initial draft of the Licensing Topical Report, a multi-disciplined review was performed by a PORC Subcommittee. Next, the full PORC reviewed the PUP, and then a review was performed by the PP&L SRC (first a subcommittee review followed by presentation to the full SRC).

The engineering evaluations for power uprate presented an opportunity to review much of the original design bases of SSES. The knowledge base and in many cases the analytical methods applied to the reviews were more mature than those applied to development of the original design bases. As a result, PUP reviews resulted in seventeen Engineering Deficiency Reports (EDRs). EDRs that addressed procedural issues included EDR G20068, "RHRSW Operating Procedure and the Ultimate Heat Sink Design Temperature," and EDR G20070, "RHR Fuel Pool Cooling Assist Mode and UHS Analyses." In one case, the RHRSW operating procedure did not provide accurate or adequate guidance to prevent exceeding the ultimate heat sink (UHS) maximum design temperature. In the other case, procedures and the UHS analysis did not recognize the required alignment of the RHR System to support multiple functions. Necessary procedures were revised accordingly in both cases.

Following the PUP implementation refueling and inspection outage, each unit was returned to power operations through the implementation of a PUP test program. This test program was conducted with at least the same considerations for tests and administrative controls that were used during the Start-up Test Program. Revisions were made to 39 procedures including: eight chemistry procedures; one health physics procedure; two instrumentation control procedures; thirteen operations and reactor engineering procedures; and fifteen engineering procedures. In addition, nine new procedures and five temporary procedure changes were required.

3. Design Basis Documentation (DBD) Project

The Design Basis Documentation (DBD) Project was begun in 1992 to organize the SSES design bases and supporting design information to better support engineering, maintenance and operational activities of the plant (see Sections II & VI). An engineering procedure describes the process by which the completed DBD is to be "validated." The goal of that validation is to assure that applicable design basis requirements have been validated against applicable design documents, procedures or via walkdowns of the physical plant. Some design basis statements describe operating conditions that need to be part of an operating procedure. Those design basis statements were validated against the appropriate procedure, or in some cases against Technical

Specifications. For example, one High Pressure Coolant Injection (HPCI) System design basis statement includes the requirement for remote-manual control of flow. The DBD Verifier refers back to the HPCI operating procedure to verify that the manual control capability is available to the operator.

4. **Improved Technical Specifications (ITS) Project**

The Nuclear Department is currently in the process of converting the existing Technical Specifications to the Improved Technical Specifications (ITS) and Technical Requirements (see Section VI.D.2b). This effort impacts a large number of Department Procedures. It is currently scoped at close to 2300 procedure changes with about 1700 affecting surveillance test procedures.

The surveillance test procedure revisions will undergo the technical review and approval process as defined in the surveillance test program NDAP. A checklist specifying review requirements and actions that must be performed is used whenever a surveillance procedure is revised. The purpose of the checklist is to assure that the requirements of the Technical Specifications and Technical Requirements are implemented. One aspect of the review specifically requires a technical evaluation to assure that the procedure, in conjunction with other procedures, completely meets the specified testing requirements.

5. **Emergency Operating Procedures (EOPs)**

PP&L has implemented Revision 4 of the Boiling Water Reactors' Owners Group (BWROG) Emergency Procedures Guidelines (EPG) at SSES, which has received evaluation from the NRC via a Safety Evaluation Report (SER). The NRC requires utilities choosing to implement the EPG to assure that implementation does not impact the design bases. PP&L implementation has been consistent with this directive, and this activity serves as another vehicle to assure design basis consistency. This activity was completed by a joint team of engineering and operations personnel. It demonstrates the direct organizational integration of the design authority (e.g., engineering staff) with the Emergency Operating Procedures (EOPs).

The process steps used to assure design basis consistency when implementing the EPG are described below.

1. Translation of the BWROG EPG into the SSES EPG: Translation of the BWROG EPG into the SSES EPG was performed in accordance with a Nuclear Department procedure. The procedure delineates when deviations from the generic guideline are allowed, one being designs that differ from that presumed in the generic guideline. PP&L has taken deviations from the BWROG EPG when the instructions are inconsistent with SSES design basis. Examples of such deviations include elimination of the ADS inhibit step from the reactor pressure vessel control procedure, and elimination of primary containment venting within the plant design basis. The SSES EPG also identifies operator actions, such as initiation of drywell sprays which are not credited by the licensing bases. In cases such as this, analysis was performed to verify that these actions do not violate the design basis. Therefore evaluation and preservation of the SSES design basis is an integral part of the development of the SSES EPG.

2. Translation of the SSES EPG into EOPs and EOP Bases: The next step in the EOP development process represented a translation of the strategy identified in the SSES EPG described above, into the EOPs and EOP bases issued to the control room. The SSES EPG is the sole basis for the SSES EOPs; therefore the EOPs reflect the design and licensing basis consistency of the SSES EPG. Additionally, when developing the EOPs and associated bases, actions necessary to comply with the licensing bases were incorporated.

3. Verification: Once the EOPs were completed, they are verified to ensure that they reflect the SSES EPG. This verification process is performed both by licensed operators and engineers and is performed in accordance with a Nuclear Department procedure. Verification is an independent review intended to ensure that: the SSES EOPs are technically correct in terms of the plant design and satisfying PP&L's defense-in-depth standards; there is a correspondence between plant procedures and plant hardware; and the EOPs correctly reflect the SSES EPG. Any discrepancies are documented and the EOPs are revised accordingly. Therefore the procedures issued to the control room are consistent with the plant licensing and design basis.

4. Validation of EOPs in the Simulator: Prior to issuing the EOPs for eventual use in the control room, the EOPs are validated in accordance with a Nuclear Department procedure. The validation process consists of observing operators responding to scenarios designed to test the EOPs. The observation is carried out by licensed operators, and engineers and occurs in two steps. First, the scenarios for dynamic simulator exercises are observed real-time. Then, a video is reviewed and the participating operators are interviewed to ensure that they fully understood the actions intended by the procedures when responding to the scenario. This validation ensures that the EOPs are usable; that the procedures will work as intended; and that procedures are compatible with staffing. Deviations from expected response are evaluated, and the EOPs are changed as necessary.

5. Preparation of Safety Evaluations: The final step in EOP preparation is the preparation of Safety Evaluations in accordance with 10 CFR 50.59. These Safety Evaluations document that EOPs do not create the potential for accidents outside the plant licensing basis. The Safety Evaluations are reviewed and approved by PORC prior to issuing the EOPs. Additionally the SRC provides a high level review to assure integrity of the EOP preparation process.

In addition to these steps, validation of the simulator was performed to ensure simulator fidelity. Therefore the fidelity of the simulator is an integral part of EOP validation. Many studies have been performed to confirm the simulator's fidelity with actual plant operation. These include the original simulator acceptance testing, which compared the simulator's response with actual plant transients, or in absence of actual plant transient response to qualified reactor transient calculations.

Calculations to verify simulator fidelity have continued beyond the initial acceptance testing. Examples include flow effects on the fuel zone Level instrument, evaluation of the new simulator's level instruments response to a large loss-of-coolant-accident, and the paper reported in NUREG/CP-0132, "Simulator Bench-Marking Studies for ATWS Scenarios." Simulator bench-marking continues as needed to support simulator fidelity. Therefore, there is a adequate confidence that the simulator provides the level of fidelity necessary to ensure that the actions identified in the EOPs, including those required by the licensing basis, will be executed as assumed in calculations, and that situations will not arise that will

result in the operator performing steps inconsistent with the plant licensing bases.

E. *Conclusion*

We have reasonable assurance that design bases are translated into operating, maintenance and testing procedures. This assurance is based upon the confidence we have in our original translation of the design bases, the confidence we have in our existing quality assurance change processes, and the ongoing QA audits and self-assessments as well as individual projects which have served to verify the translation.

IV. Response to Item (c): Consistency of Actual Plant Configuration and Performance With Design

A. Introduction

1. Purpose

This section of the response provides PP&L's:

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

2. Overview

PP&L's processes for design and configuration control of the Susquehanna Steam Electric Station (SSES) were established to meet the applicable requirements of 10CFR50, Appendix B. These requirements are currently established in the PP&L Operational Quality Assurance (OQA) Program contained in Chapter 17 of the Final Safety Analysis Report (FSAR) and the Operational Policy Statements (OPS).

Section II of this response provides the detailed discussion of the original design and construction programs which established the original plant design "baseline" and the testing program that confirmed the translation of this baseline into actual plant configuration. Section II also discussed details of the processes in use at SSES to assure that impacts on the design margin are identified and addressed. Section III described the processes used to translate both the original design baseline and design changes into plant procedures. This section builds upon those discussions to establish how the Initial Testing Program coupled with results of audits, assessments, inspections and tests, provide reasonable assurance that the design bases have been accurately translated into plant configuration, as well as how current processes continue to assure proper translation.

[illegible]

As described in preceding sections, PP&L took an active role in the original design and construction of the plant to ensure that work was performed in accordance with appropriate quality assurance principles and controls. Throughout start-up, PP&L was involved in the evolution of the processes controlling key activities. Therefore, by the time PP&L assumed full control of the design and configuration of SSES, PP&L personnel had accrued substantial experience with the design and physical plant. This involvement also resulted in an appreciation of the design and configuration control processes which allowed PP&L to assume that final responsibility via a smooth transition from the original designer. Design and configuration control continued forward, under the PP&L program, which was designed to allow changes to be made to an "operating" plant.

3. Organization

The detailed response to item (c) addresses the key points listed below.

- Consistency of Original Design With Plant Configuration and System, Structure, and Component (SSC) Performance
- Processes for Ensuring Consistency of Plant Configuration With Design Bases
- Activities Which Confirm Consistency of Plant Configuration with Design Bases
- Activities Which Confirm Consistency of Plant Performance with Design Bases

B. Consistency of Original Design With Plant Configuration and System, Structure, and Component (SSC) Performance

As described in Section II, PP&L was in substantial control of the design and construction of SSES in the role of overseeing the design/construction programs and activities. Following construction, PP&L assumed control over systems, structures and components through a formal turnover from the construction organization to PP&L, which involved detailed walkdowns by Bechtel Construction, PP&L Construction, and the Integrated Start-up Group personnel to verify both construction status, as well as conformance of the plant configuration with design output documentation. Regarding SSC performance, the Initial Test Program,

described in FSAR Section 14, confirmed the system/functional design bases identified in the SAR, and demonstrated, to the extent practicable, that the plant was capable of responding as designed to anticipated transients and postulated accidents.

The acceptance criteria section for each preoperational/acceptance test/start-up test procedure identified the criteria necessary to determine that system performance was acceptable. Each listed criteria, reference sources (see Section II), and the steps in the text of the procedure that verified the criteria had been satisfied.

FSAR Section 14.2.5 details the review, evaluation, and approval process associated with the test results. Responsibilities and authorities were vested in the Test Director, the Group Leader, the Test Review Board, Plant Operations Review Committee, and the Plant Superintendent. Additionally, an independent review of the test results was performed by the QA staff as an additional assurance as to the completeness and adequacy of the testing. The QA group also verified resolution of any exceptions.

During the process of check-out, initial operation, and preoperational or start-up testing, design issues were encountered. These design problems were formally documented and reported to the appropriate design organization for resolution. A response from the design organization for such reported items was mandatory. If the response required a facility modification, the modification was installed and tested to confirm problem resolution, and the appropriate design documents were revised and distributed to controlled files. Once again, these activities were conducted under QA controls and practices responsive to 10 CFR 50, Appendix B.

The detail to which the Initial Test Program was conducted and the results obtained from the checkout and testing give us a adequate confidence that the original design and construction activities adequately translated the design bases into the plant configuration, and that SSC performance was consistent with design bases.

C. Processes for Ensuring Consistency of Plant Configuration With Design Bases

One reason for our confidence that SSES design bases continue to be accurately and consistently reflected in the actual plant configuration, is an effective set of design controls that reflect our quality management philosophy. This configuration control philosophy is an integral part of our activities and change processes, and holds that we will: (1) not knowingly make changes to the plant without understanding how the

change affects the design bases; (2) make those changes via the correct process; and (3) update the impacted managed information sources.

Managed information sources, which contain plant configuration information, are available in either an as-engineered or as-built status. Such information sources are provided with a change mechanism process that assures a level of review and approval suitable to the specific information source, as well as procedures which establish the information update requirements such that the updated information is available upon approval and issue. Information sources made available to users in an as-built condition are controlled such that the as-engineered status is available and tracked and the as-built condition is incorporated only after the actual change has been made.

As described in Section II, the configuration management requirements are an integral part of the plant modification program. As such, each change to the plant provides a new opportunity to evaluate a portion of the plant design bases. The modification process itself is designed to assure that the design bases are identified, are properly converted into the design, are reflected in design output documents and then into the actual plant configuration. Section II provides details of how the modification processes assure identification and translation of design bases into the plant configuration.

D. Activities Which Confirm Consistency of Plant Configuration with Design Bases

PP&L has a number of on-going programs and has had projects in the past whose results provide added confidence that the actual plant configuration is consistent with the design bases. Some examples are discussed in the following subsections.

1. Plant Modifications

From January 1, 1984, approximately 4400 modifications have been completed. These include both major and minor modifications, primarily Design Change Packages (DCPs), Engineering Change Orders (ECOs), and Setpoint Change Packages (SCPs). These also include modifications associated with general major projects (e.g., Power Uprate Program, E Diesel Generator). Each of these required the identification of the design bases, from various available sources to support the modification,

and the physical work involved in implementing the modification. Thus, for each modification, the design bases and the plant have been compared through the design and installation of the modification itself. Virtually every system in the plant has been touched by the modification process.

2. DBD Preparation and Validation

The ongoing Design Basis Documentation (DBD) Project was begun in 1992 to organize the SSES design bases and supporting design information to better support engineering, maintenance and operational activities of the plant. To date, 18 completed DBDs have been issued. The scope of these DBDs includes many of the primary safety features of the plant. See Section VI for details of DBD Project, including how DBDs are developed, validated and how open items are processed.

An engineering procedure describes the process by which the completed DBD is "validated." The goal of that validation is to assure that applicable design basis requirements (as defined in NUMARC 90-12) have been identified and that a representative sample of those design basis requirements have been "validated" against applicable design documents and plant configuration information. In fact, most of the design basis statements in the DBDs are validated.

As a result of the substantial effort to develop the first 18 DBDs, approximately 390 open items were documented for further study and resolution. To date, less than one dozen discrepancy reports have been issued based on DBD project activities. None of these represented an operability or reportability concern.

3. Safety System Functional Inspection (SSFI) Activities Which Confirm/Assure Consistency

Three SSFIs have been performed since mid-1988. An SSFI is a specialized inspection to determine whether a single safety system and its supporting systems, as designed, installed and configured is capable of performing its safety function. It is an in-depth, multi-disciplined, highly technical review of a very narrow scope of the plant, usually a system. The inspection begins with a detailed review of the design and the design bases, and the modification and configuration control programs. Other team members evaluate

operations, maintenance and QA activities associated with the system. The SSFI is less rigidly structured than other types of inspections like audits, in that, if during the SSFI a weak area is discovered, emphasis will be placed on that area. The weak area is probed by more than one team member until the entire problem is uncovered. Because of its structure and staffing, there is a high level of confidence that the results of an SSFI present an accurate picture of the system.

- Safety System Functional Inspection (SSFI) of the Emergency Service Water System

  In 1988, PP&L initiated a Safety System Functional Inspection of the Emergency Service Water (ESW) system. The scope of the inspection included the ESW System and certain safety-related systems which support ESW operation, including Engineered Safeguards Service Water (ESSW) Pumphouse ventilation, the 480-volt and 120-volt AC systems, the 250-volt and 125-volt DC systems and the emergency diesel generators. The inspection was performed using the guidance delineated in Chapter 2515 of the NRC Inspection and Enforcement Manual. One of the inspection requirements was to determine if the as-built configuration of the systems was consistent with the current design bases. The inspection was completed over a period of eight weeks by a team of eight independent reviewers, assisted by several PP&L individuals.

  The team concluded the ESW system to be functional in accordance with the criteria detailed in NRC IE Manual, Chapter 2515. The system was found to fulfill PP&L's regulatory commitments and would operate as expected in response to an accident. The team issued a total of 33 observations, including the susceptibility of ESW flow transmitters to flooding, missing remote position indication testing, the need to account for instrument error in the spray pond temperature, and ESSW Pumphouse ventilation issues. Corrective actions included a few procedure changes, a few calculation improvements, several maintenance and testing activities and one modification to replace the Diesel Generator nameplate.

- Electrical Distribution System Functional Inspection (EDSFI)

NRC SSFI #90-200 was performed by the NRC in August 1990 on the Electrical Distribution System to determine if the electrical distribution system would be capable of performing its intended safety function as designed, installed and configured. The plant electrical distribution system is perhaps the most widely distributed system in the plant, touching virtually every portion of the plant.

The inspection was performed by an NRC inspection team consisting of six NRC personnel and three NRC consultants. The team was on site for two full weeks. PP&L began preparing two months prior to the inspection with a team of individuals. At the height of the project, PP&L devoted approximately 50 people to this effort to gather information for, and respond to questions from, the inspectors.

The inspection report noted strengths in the areas of coordination among corporate engineering, the site and licensing; control of instrument setpoints; the quality of modification engineering; and thorough QA Audits. Weaknesses noted were in the discrepancy management program and the lack of overpressure protection in the Diesel ESW piping. In addition there were 12 other findings that required equipment or analysis work to resolve. (See Section V regarding the evolution of engineering discrepancy management systems.)

The inspection team concluded that generally the SSES electrical distribution system would be capable of performing its intended safety functions. With the exception of the 14 specific findings identified in the report; the batteries, emergency diesel generators, switchgear, and other components within the Electrical Distribution System were found to be adequately sized and configured. Separation between redundant trains or divisions was found to have been adequately maintained, and an adequate design basis existed and was being upgraded and maintained for the SSES.

The PP&L team remained intact for some period of time after the inspection to address and close all the open items that had been raised. This inspection was a very detailed comparison of the consistency of the design bases with the actual

configuration of the plant. With the improvements made after the inspection, we have adequate confidence in the consistency between the design bases and the configuration of the electrical distribution system.

- HPCI SSFI

An SSFI of the High Pressure Coolant Injection (HPCI) system was performed in August, 1992. The inspection was performed by an independent team of five external individuals with SSFI experience, with the support of the PP&L QA organization.

Relevant results of the inspection included strengths in the areas of technical staff capability, the MOV (GL 89-10) Control Program, and HVAC design documentation. Three potential weaknesses were noted in the areas of the test line to condensate storage tank (CST) valves, calculation control and Unit 1/Unit 2 coordination.

The inspection resulted in two Engineering Deficiency Reports to track completion of actions and to initiate further evaluation. One of the two EDRs involved a question of the adequacy of the CST volume reserved for HPCI. The other involved a HPCI mode for which certain valve loads had not been considered in the 250-volt DC battery load profile. The CST volume proved to be adequate and the HPCI mode in question required a Technical Specification change to revise the battery load profile. A revised calculation control program has been put in place. None of the issues raised by this inspection represented an operability or reportability concern. The SSFI team concluded that the system was adequately designed and installed, and would operate as designed under postulated design basis accident conditions.

- Service Water System SWSOPI Readiness Inspection

In addition to the three SSFIs noted above, in March 1995, PP&L conducted an inspection of the Emergency Service Water system in preparation for an NRC Service Water System Operational Performance Inspection (SWSOPI).

The Emergency Service Water (ESW) and the Residual Heat Removal Service Water (RHRSW) systems were the two primary systems covered by this review. The focus of this

inspection was to assess PP&L's response to Generic Letter 89-13. The inspection team was composed of four reviewers, was on site for four weeks, and independently reviewed and assessed documentation, and where possible, activities which detailed the items listed below.

-- the current design integrity of the ESW and RHRSW systems to perform thermally and hydraulically, as required during an accident event;

-- the adequacy of surveillance and testing practices being performed for their ability to accurately represent the current condition of the systems and their components and predict when performance degradation below acceptable values may occur; and

-- the acceptability of the present plant operations and maintenance practices and procedures to adequately control the operation and line-up of the ESW, RHRSW, and adjoining systems, and perform the appropriate corrective action when required.

Strengths were noted in the areas of comprehensive design documentation; the existence of flow models for ESW and RHRSW; personnel qualifications and organizational strengths; and an effective plant maintenance program. Weaknesses were identified in the areas of the basis for the heat exchanger fouling factors, and references in operating and maintenance procedures. The team issued 17 observations of potential concern. All were evaluated, and no effects on operability were identified. Actions to close the items continued after the inspection report was issued. Two CRs were issued as a result of this inspection: one to address that fact that the ECCS room coolers used an air-side fouling factor that had inadequate basis; and the second to address a tube-side fouling factor on the "E" Diesel Generator Lube Oil Cooler. The fouling factor effect was evaluated and found to not affect the operability of the coolers. This inspection raised no issues that were determined to affect system operability.

Based on the degree to which these inspections touched a wide range of the plant, and the lack of nuclear safety significant findings from these inspections, we conclude that there is reasonable assurance that the systems in question are capable of performing consistent with established design bases.

4. QA Activities

Activities performed throughout the design, procurement, construction, turnover, testing, and operational phases of SSES have been performed under a QA Program that is responsive to 10 CFR 50, Appendix B. This program set forth the QA policies and practices that would be requisite throughout the life cycle of SSES. Since the beginning of the initial design, PP&L has had an independent QA/QC staff made up of PP&L employees that provides an oversight, inspection and evaluation function. Through the implementation of audits, surveillances, inspections, document reviews, event reviews, and participation in construction and engineering turnovers, the QA staff has been able to furnish engineering and plant management with reasonable assurance as to the consistency of the plant configuration with the design bases.

PP&L's QA audit program is responsive to 10 CFR 50, Appendix B. The planned and periodic audits provide a mechanism to evaluate the adequacy and effectiveness of the QA program and practices in assuring compliance with PP&L policies and rules, approved design bases, licensing commitments, operating parameters and philosophies, and the various administrative management processes established to assure compliance. Periodic audits of the programs and processes established for design control, configuration management, procedure development and turnover and control of plant equipment and systems, conduct of plant maintenance, procurement controls, testing, and plant operations have been, and continue today, to be conducted and used to evaluate the health of the established quality assurance programs and practices.

From 1982 through 1996, PP&L's QA organization has conducted over 400 internal QA audits covering over 40 programs and process areas. The formal QA audit program has evolved over the life of SSES and today is supplemented by a broad department-wide assessment program that includes both independent assessments and self-assessments as measures of continued compliance with quality assurance practices. These assessment results are another measure of assurance as to the level of consistency between the plant configuration and the design bases.

In addition to the audit program, the QA staff participated in system and equipment turnovers, system walkdowns, and start-up testing activities that assured the establishment of a solid baseline for both the design bases and plant configuration. Subsequent to the establishment of the baseline, through the conduct of QA surveillances, QC inspections and monitoring, in-line reviews of design documents and plant procedures, and participation in SSFI type assessments, the QA staff has been able to evidence continued consistency between the plant configuration and design bases. Once again, historical data indicates generally adequate establishment and implementation of appropriate quality assurance practices.

5. Other Reviews and Walkdowns

There are other reviews and walkdowns, both on-going and one-time projects, which help to confirm the consistency between the design documentation and the plant configuration. No one of these activities covers the entire plant, but taken together, a substantial portion of the plant is reviewed.

a) Individual Plant Evaluation (IPE)

The NRC issued Generic Letter 88-20 requesting that each licensee perform an examination of facility vulnerabilities to severe accidents. The PP&L IPE was written to address that request. The analyses that were required to support creation of the IPE required an accurate understanding of the plant configuration. The information gathering for the IPE included walkdowns, on an as-needed basis, to verify the as-built configuration of portions of the plant. For example, a primary containment walkdown was performed to assess as-built plant configuration information in support of creating the computer model of the primary containment.

b) Individual Plant Evaluation for External Events (IPEEE)

The IPEEE involved plant walkdowns to verify seismic equipment was properly anchored per the design, to locate unacceptable seismic interaction between equipment, to

locate fire sources and to identify areas susceptible to damage from wind and flooding.

The seismic portion of the IPEEE provided a limited verification of the seismic design basis for the safety-related equipment associated with two safe shutdown paths. Specifically, the IPEEE involved plant walkdowns that verified the specific equipment models that were qualified and that the equipment was properly anchored as per the design. The walkdowns identified seismic interaction concerns that were present that could possibly jeopardize the original equipment qualification. During the walkdowns, the equipment's physical condition was reviewed to assure that it was in conformance with the equipment's qualified arrangement.

c) *10 CFR 50, Appendix R*

10 CFR 50, Appendix R requires a fire protection program, a fire hazards analysis, and fire prevention features. The original safe shutdown and fire hazards analysis was completed by the plant designer prior to plant operation. In a 1985 inspection, the NRC concluded the Fire Protection Program did not conform to the NRC's interpretation of Appendix R. By the end of 1987, PP&L had completed a safe shutdown reanalysis and concluded that with the implementation of identified modifications, the plant would be in full compliance with Appendix R. The safe shutdown analysis was revised in 1995/1996 in support of resolving concerns that surfaced about the Thermo-Lag 330-1 material. These analyses have resulted in the design bases collected and compared to the design several times over the years in controlled analyses. The associated in-plant work has compared the design with the plant configuration.

The results of the Appendix R work performed over the years provide confidence that the affected plant configuration has been maintained consistent with the applicable design bases.

d) *Power Uprate Program*

The Power Uprate Program (PUP) was begun in 1991 with the objective of increasing the power output of both SSES units by approximately 5%. The effort included studies to review each affected plant system to determine the system's ability to support the uprated conditions. The review entailed evaluation of the original design requirements and bases for the system, the current plant operating conditions, and the conditions which would be expected to exist at the uprated power condition. As a result of these reviews, recommendations were made for modifications, testing requirements and revised operating procedures to accommodate the uprated conditions. The PUP reviews touched most of the systems in the plant.

Additionally, PP&L arranged an independent assessment consisting of a technical and programmatic review of activities in support of the power uprate. At the time of the assessment, the majority of the engineering analyses had been completed. The assessment team conducted the assessment through a review of documents prepared to support PUP efforts, personnel interviews, observations, and some system walkdowns. The assessment team found no hardware or operational issues that would prevent SSES from achieving the proposed uprated conditions.

Refer to Section VI provides for additional details on the Power Uprate Program.

e) *Walkdowns*

Over the more than 13 years since Unit 1 went into commercial operation, a number of activities have included walkdowns of one or both units that, to varying degrees of detail, formality and scope have compared the design with the physical configuration of the plant. Examples of these activities are:

- During the 1980s, a project was completed to compare selected data in the Susquehanna Equipment Information System (SEIS), a computer database, with actual plant equipment. The data that was a part of SEIS is now in

the Nuclear Information Management System (NIMS), a developing Department database application.

- Walkdowns were conducted to identify and document the manufacturer, type and rating for Class 1E Control Fuses. The results were made a part of the Fuse Control Program. Similar walkdowns were conducted to identify the manufacturer, model and size of each Class 1E, 250-volt DC and Class 1E, 480-volt AC overload heater in the scope.

- System Engineers perform periodic walkdowns of the systems assigned to them to assess the material condition of the system. These walkdowns are usually informal, but are done by knowledgeable people looking for anomalies.

6. Activities Which Confirm Consistency of Core Configuration

The responsibility for implementation of reactor core reload design changes resides with the SSES plant staff. Specifically, the on-site Reactor Engineering group is responsible for verifying that the nuclear fuel in each reload core is arranged consistent with the cycle-specific reload design. This is accomplished by verification of each reload core using the reload cycle-specific Core Loading Map which supported the PORC approved reload cycle specific Safety Evaluation.

Activities which confirm/assure consistency of the actual reactor core configuration with the design requirements occur on a routine basis during and following each reactor refueling. These activities are controlled through a combination of Nuclear Fuels Functional Unit Procedures (NFPs) and SSES Operations Functional Unit Procedures (RE procedures). The Reload Design and Analysis Program establishes the controls for documentation, review, and approval of reload cycle-specific core design configuration information. It identifies controls for the transmittal and confirmation of receipt of this design configuration information between Nuclear Fuels and the SSES Plant Staff. Specifically, transmittal and receipt of the reload cycle-specific Core Loading Map and Control Blade Change Specification are addressed in this NFP. The Core Loading Map and Control Blade Change Specification are utilized in establishing the change process for the

reload core in accordance with written procedure. Independent verification of the reload core configuration is performed using the Core Loading Map and the core/fuel pool verification procedure. The as-loaded control blade configuration is transmitted to Nuclear Fuels for verification of consistency with the Control Blade Change Specification.

No one of these activities confirms the consistency of the entire plant, but each looks at and confirms the consistency of a portion of the plant configuration with the design documentation. Any time this happens, the possibility of finding inconsistencies is present. Each provides an opportunity to identify apparent differences between the design information and the actual plant configuration. These can be entered into the Apparent Configuration Discrepancy (ACD) process for evaluation (see Section V) or into the Condition Report process directly. These differences are evaluated and any necessary steps, analysis or modification, taken to assure that the design bases, design documentation and the actual plant configuration match.

E. Activities Which Confirm Consistency of Plant Performance with Design Bases

Many mechanisms are utilized to assure that the actual plant performance is consistent with the plant design bases. Such mechanisms include those that test or monitor actual on-line plant performance as well as those that utilize the plant simulator to predict plant performance.

Testing occurs following plant modifications and maintenance activities, or as required by routine surveillances to demonstrate system operability in accordance with the plant's Technical Specifications, and/or system performance in accordance with the plant design basis. System performance monitoring occurs in conjunction with the Nuclear Maintenance preventive maintenance programs as well as Systems Engineering system reviews.

In addition to formal programs for testing and monitoring, the routine continuous operation of the plant by on shift operations personnel serves as a mechanism for assessing performance in accordance with the design basis. Any abnormal performance observed results in the generation of a Work Authorization (WA) to investigate, and/or identification of the problem within the corrective action program. This provides for evaluation of the problem for significance and the formulation and implementation of corrective actions to assure positive closure. (See Section V).

1. System, Structure and Component Testing/Monitoring

Throughout the life of the plant, various kinds of testing is performed to assure that altered systems or components perform as required, and that the unchanged systems and components continue to meet the performance requirements of the as-built design. No one test is capable of verifying the performance of all the plant systems. However, taken together, the testing that is done at the plant provides a sufficiently broad picture that we have reasonable assurance that the plant is performing consistent with the design requirements. Any specific results which show inadequate performance are resolved via the appropriate process.

a) Surveillance Testing

The Surveillance Testing Program establishes the administrative controls for implementation and maintenance of the surveillance procedures and tests used to satisfy the requirements identified in the Susquehanna Technical Specifications. There are over 900 individual Unit 1 and common surveillance procedures in eight categories. The categories include chemistry, engineering, health physics, instrumentation, maintenance, operations, technology and reactor engineering. The surveillance procedures have established the specific acceptance criteria necessary to confirm operability of a given plant component, subsystem, or system. The program assures that the testing is performed:

1. At the frequencies described in the Technical Specifications.
2. Prior to changing Reactor modes in which the applicable equipment is required.
3. Prior to declaring a Technical Specification or Technical Requirements related system/component operable, after applicable maintenance activities have invalidated the current surveillance of record.
4. Prior to declaring a Technical Specification or Technical Requirements related system/component operable, after applicable modification activities have invalidated the current surveillance of record.

5. Prior to declaring a newly installed Technical Specification or Technical Requirements related system/component operable.

b) *Post-Maintenance Testing*

Post-maintenance testing also includes both functional, performance and operability testing in order to verify that component system operation is as expected. Identification of testing is the responsibility of the Nuclear Maintenance work group performing the work, with the assistance of Nuclear Systems Engineering, as needed.

c) *Post-Modification Testing*

Testing is performed following installation of a modification to show that the system, in its new configuration, performs in accordance with expected performance which is per the design basis for the system. Testing consists of: (1) functional testing, to assure that the modified component(s) are functionally sound an ready for operation; and (2) operability testing, to show that the structure, system or component is capable of meeting minimum operability requirements of the plant's Technical Specifications, Technical Requirements and Performance Testing to show that the system/components are operating within the original design envelope. Identification of functional testing is the responsibility of Nuclear Modifications or Nuclear Maintenance. Identification of operability and performance testing is the responsibility of Nuclear Systems Engineering.

d) *Leakage Rate Test Program*

The Leakage Rate Test Program implements the administrative controls for the primary containment leakage rate testing as required by 10 CFR 50 Appendix J, Option B and is in accordance with Regulatory Guide 1.163 and NEI 94-01. The governing procedure covers bypass leakage requirements and reactor coolant pressure isolation valve testing. The primary containment leakage rate testing program is applicable to every primary containment boundary component and the primary containment structure.

The procedure contains the requirements for complying with Option B, including administrative limits and performance-based testing frequencies, tagging requirements, flowpath, documentation of evaluations for foreign potential, investigation guidance, and review requirements for LLRT packages. Additionally, this procedure outlines the requirements for measuring secondary containment bypass leakage, CRD Seismic Island check valve leakage, the drywell-to-suppression chamber bypass leakage, and the Control Rod Drive Header Leakage.

e) *Station Pump and Valve Testing Program*

The Station Pump and Valve Testing Program specifies the administrative controls which implement the ASME Section XI Pump and Valve Testing Program and specifies the specific SSES pumps and valves that are subject to this program. The governing procedure specifies requirements for general inservice testing, pump and valve program development, specific pump and/or valve testing, test scheduling, instrumentation, test performance data analysis and test records.

f) *ASME Section XI System and Component Pressure Testing Program*

The ASME Section XI System and Component Pressure Testing program establishes the administrative controls

necessary for implementation of those portions of the Nuclear Department Inservice Inspection (ISI) Program relating to system pressure testing at SSES. The scope of the program encompasses the system pressure testing requirements of Section XI of the ASME Boiler and Pressure Vessel Code, Article IWA-5000 as applied to SSES by 10 CFR 50.55a and Technical Specifications. This includes component pressure testing required by post-maintenance testing of ASME Code components.

The governing procedure specifies the general in-service inspection requirements, general pressure test requirements, pressure tests of the Class 1 boundary, pressure tests of the Class 2 systems, pressure tests of the Class 3 systems, VT-2 examination requirements, non-VT-2 examination requirements, and records requirements.

g) Emergency Diesel Generator Reliability Monitoring Program

The Emergency Diesel Generator Reliability Monitoring Program, establishes the requirements necessary for operating and maintaining the Emergency Diesel Generator Reliability Program in accordance with NUMARC 87-00, Revision 1. The Emergency Diesel Generators credited in the station blackout coping assessment are required to be maintained at or above a specific target reliability of 0.975 (97.5%). The governing procedure specifies the methodology for monitoring diesel generator reliability, maintenance of the reliability and corrective maintenance data, evaluation of the performance and reliability indicators, comparison of the reliability indicators to the target values, remedial actions including root cause assessment, and reporting requirements to the NRC. During 1996, the Diesel Generator combined availability was 99.24%, and the start reliability was 100%.

h) Motor Operated Valve (MOV) Program

The Motor Operated Valve (MOV) Program defines PP&L's response to the requirements of the NRC Generic Letter No. 89-10 "Safety Related Motor-Operated Valve Testing and Surveillance." The program provides that the design bases are documented for each valve within the scope of GL 89-10. It also provides for identification of the testing, inspection and maintenance of MOVs so as to provide the necessary assurance that they will function when subjected to the design basis conditions that are to be considered during both normal operation and abnormal events within the design basis of the plant. In order to ensure continued functionality of these MOVs, a trending and periodic verification program is being established per the recommendations of GL 96-05 "Periodic Verification of Design Basis Capability of Safety-Related Motor Operated Valves."

i) Heat Exchanger Program

The Heat Exchanger Program defines the responsibilities related to the operation, cleaning, maintaining, inspecting, monitoring and design of SSES Heat Exchangers. This program identifies the various functional unit responsibilities in maintaining the performance of the SSES heat exchangers. The program provides ongoing assurance that plant heat exchangers performance is consistent with plant design,

j) Nondestructive Examination Program

The Nondestructive Examination Program provides guidance on the responsibilities for the implementation of nondestructive examination (NDE) requirements under the Operational Quality Assurance Program, as mandated by 10 CFR 50.55a and the ASME Boiler and Pressure Vessel Code, Section XI. There are a number of organizations within the Nuclear Department involved in accomplishing the various facets of NDE that are required to support the needs of SSES.

k) Fire Protection Program

The Fire Protection Program outlines the responsibilities and actions required to implement the Nuclear Department Fire Protection Program. The Fire Protection Program includes: fire prevention, passive fire scope barriers, fixed fire suppression systems, etc. The program assures that the fire protection features comply with commitments, including the Fire Protection Review Report and appropriate provisions of 10 CFR 50, Appendix R, and that there is adequate testing to assure the fire protection features continue to perform consistent with the requirements.

2. Simulator

During the late 1980's, 10 CFR 55 required that, after May 26, 1991, operator testing could only be performed on a "certified" control room simulator. PP&L's existing simulator required a major upgrade to be certifiable.

The SSES simulator contains the Unit 1 and common control room panels and has sufficient panels to conduct ANSI/ANS 3.5 - 1985 evaluations and malfunctions applicable to SSES. The new simulator required more detailed system models, including more sophisticated response to plant transients. During development of the new simulator models, each model (in particular, the core, recirculation system and containment models) were evaluated against both design basis information (the "best estimate" response to design basis events, since the simulator is designed to give the operator the feel of the actual plant performance) and actual plant transients to assure that the certification requirements of ANS/ANSI 3.5 - 1985 could be met. In addition, to continue to conform to the certification requirements, the simulator undergoes a complete benchmarking of all design basis and operational transients, as well as any new actual plant transients every four years. This level of fidelity has allowed the use of the simulator to observe the impact of a change on system or plant response and for use in the trial operation of procedures.

3. Equipment Performance Trends/Indicators

PP&L has a number of data sources and indicators to use as part of the ongoing evaluation of plant performance. The scope of trended data includes equipment, system and plant performance data associated with program and process implementation. Trends adverse to quality or reliable operation are identified and can then receive the appropriate level of management attention for corrective action. Once the corrective action is implemented, the same data trending is used as an indicator to assure that the corrective actions bring about desired results. Listed below are a number of data sources and trend reports we use. This is not necessarily an exhaustive list, but is indicative of the information available.

- Condition Report (CR) Trending - The CR database contains trend coding, including event codes and cause codes, allowing for various types of trend analyses.

- Independent Safety Evaluation Services (ISES) Summary Assessment Report - A nuclear safety assessment of Nuclear Department activities, including investigations of operational incidents, performed by the Independent Safety Evaluation Services group within Nuclear Assessment Services (NAS). These reviews establish the actual response of the plant to events in comparison with the expected response.

- NAS Audit Reports - Audit Reports issued by NAS. These provide a description of audit scope, identification of the auditors, personnel contacted from the audited organization, and a summary of audit results including a description of any audit deficiencies and observations/recommendations, and a statement of the effectiveness of the quality assurance program elements that were audited.

- Safety System Performance Indicator (SSPI) - Indicator of selected SSES system availability information, issued quarterly.

- Nuclear Department Reliability, Availability, Maintainability (RAM) Report - A yearly department report that assesses SSES cycle performance with the U.S. and international nuclear communities. Operating, mid-cycle and refuel losses are analyzed.

- Component Failure Analysis Report (CFAR) - INPO report which ranks a component group failure performance of a nuclear plant against the performance of like components by the rest of the industry.

- Predictive Maintenance Periodic Report - Report of condition monitoring tasks that are performed to analyze equipment performance and detect developing degradation or abnormalities.

- Operator Rounds Sheets - Data recorded by each plant operator related to the shift activities he performed or witnessed.

- Nuclear Plant Reliability Data System (NPRDS) - INPO computer database with selected system and component failure information for domestic nuclear plants.

This listing gives an indication of the types of trending and reporting activities that are a part of the routine operation of SSES. Each of these activities has a trained, knowledgeable individual collecting and evaluating data about some function or physical area of the plant. Any anomalous situations are entered into one of the available processes (WA, CR, ACD, modification, etc.) for investigation.

4. SSFIs

Three SSFIs were performed since mid-1988. The three SSFIs and their results are described in some detail in Section IV.D. That section describes the SSFI as verifying consistency between the design bases and the plant configuration. The SSFI also examines the performance of the system relative to the expected performance presented by the design bases. As previously noted for each of these SSFIs individually, each inspection concluded the system under consideration was capable of performing its safety functions within the design bases set for them.

In addition to the three SSFIs described above, PP&L conducted an inspection of the Emergency Service Water system in preparation for an NRC Service Water System Operational Performance Inspection (SWSOPI), also described in Section IV.D. As noted before, this inspection raised no issues that were determined to affect system operability.

As noted in Section IV.D, after each of these inspections, a focused effort was made to strengthen items determined by the inspection to be weaknesses. These focused efforts should preclude recurrence of the weaknesses identified. Based upon the degree to which these inspections touched a wide range of the plant, and the lack of nuclear safety significant findings, we conclude that there is reasonable assurance that the systems in question are capable of performing consistent with established design bases.

5. System Performance Focus: System Engineers & System Reviews

Nuclear System Engineering and the individual System Engineers are the focal point for plant system support and analysis at SSES. The mission of Nuclear Systems Engineering is to optimize plant systems throughout the life of the plant in support of overall plant performance objectives and department mission. The System Engineers monitor and evaluate the status of their systems, assess system performance, status and potential changes in light of the design basis and design intent of the system and the system's overall interaction in the plant design. They perform or direct the technical activity of the Department to assure optimum system performance in support of the Department mission.

The System Engineers accomplish these responsibilities through personal contact with operating, maintenance and modifications personnel, review of various types of documentation and logs associated with those functions, and access to various types of design and performance information. Using this information they make recommendations to maintain or improve system and plant performance. They also develop other testing to validate changes to the plant and to collect needed information not collected otherwise.

One of the Systems Engineer's responsibilities is to issue a periodic System Status Report (SSR) for selected systems. The System Engineer's routine efforts provide for regular monitoring of activities affecting the system. The SSR provides an opportunity to assemble an in-depth assessment of the system for communication with the rest of the department. The SSR consists of an executive summary which is a brief, stand-alone summary of the highlights of the SSR. Other information provided in the SSR includes: (1) system performance, to discuss availability, trends associated with system performance, areas of concern, functional failures, and applicable industry events; (2) material condition, to discuss results of walkdowns, equipment out of service, and trends associated with material conditions; (3) deficiencies, to discuss progress on old items, new items identified and trends of deficiencies; and (4) improvements, to discuss modifications, procedure improvements, improvements to maintenance strategies, future enhancements, and activities to improve single failure reliability. Areas like these, and others, are discussed as appropriate to the system and time period in question.

The procedural guidance also provides for a periodic System Review Meeting attended by management members and any other parties with specific interest in that particular system. About weekly, a system is selected to be discussed at the System Review Meeting. The considerations addressed by the SSR forms the basis for the material presented at the meeting. The System Engineer presents the material in the meeting and is available for questions and discussion about any aspects of the system under review. This meeting, while somewhat informal, is an excellent forum for periodic statusing of a system and opening the floor to questions and discussion on system aspects that may not have been raised, otherwise.

The System Engineer, as the focal point for activities on a particular system, has strong ownership to assure maintenance of

configuration of a system and to identify performance issues before they become problems.

6. Reload Core Performance

The primary mechanism for ensuring that actual reload core performance is consistent with the design bases is the reload start-up tests conducted prior to, and during, the first reactor start-up following completion of each reactor core alteration evolution. In addition to the tests performed as part of the start-up following a core alteration, PP&L performs continuous comparisons throughout the operating cycle to evaluate actual core performance against predicted operation. The results of the reload start-up tests and the subsequent comparisons performed throughout the operating cycle ensure that the reload core is performing consistent with the design bases.

The following SSES activities govern execution of the reload start-up tests which are used to ensure that actual reload core performance is consistent with the design bases:

- Shutdown Margin Demonstration
- Reactivity Anomaly Check
- In-Sequence Critical and Shutdown Margin Determination
- TIP Asymmetry Check

Verification of the results of the reload start-up tests against the applicable test acceptance criteria ensures that the reload reactor core is performing consistent with the design bases. The results of each reload start-up test are documented in accordance with the applicable procedure.

The Nuclear Fuels in-core fuel management procedure defines the scope of the analyses performed as part of the ongoing core performance evaluation process. Through these ongoing evaluations, Nuclear Fuels personnel perform comparisons between core performance parameters and predicted or expected values to verify consistent performance relative to the reload design and licensing analyses. These evaluations include comparisons of core reactivity, local power distributions, TIP system asymmetries, and thermal limits.

7. Response to Transients

PP&L uses a post-event evaluation mechanism to provide ongoing assurance that the actual performance of plant systems and equipment is consistent with the design bases. Following significant plant transients, scrams, or shutdowns which occur at SSES, a thorough evaluation of plant parameters is required per the Post-Reactor Transient/Scram/Shutdown Evaluation process. This evaluation identifies anomalous plant responses and determines the cause of the event. The post-event evaluation must determine whether the equipment and systems functioned in accordance with the design bases.

Performance testing is a measure used for most of the normal plant functions to verify that the system or component is performing as expected. The area of the integrated plant response to accidents, like the Design Basis LOCA, is an area of plant performance where it is either not possible or at least very undesirable to conduct an integrated test. In these cases we depend on analytical methods coupled with known performance of the systems involved to establish the confidence that the integrated response will be as expected.

Each system upon which the integrated response depends can be individually tested to assure that its performance is consistent with the design bases established for that system. Section II discusses the Initial Test Program and Section IV.E.1 discusses the current testing programs. Testing is to assure each individual system remains well within the limits assumed in the design basis event response analysis. Using the performance of the individual systems involved, analyses are performed to determine the plant's capability to provide an integrated response to the various accidents for which actual testing is undesirable. Based on the performance of individual systems and the analysis of the integrated response, we have reasonable assurance that the plant performance remains adequate to mitigate all postulated design basis accidents.

F. Conclusion

Based upon the above, PP&L concludes that there is adequate confidence that the configuration and performance of the as-built plant are consistent with the design bases. There is also adequate confidence that the current processes and programs provide reasonable assurance that the plant configuration and SSC performance will be maintained consistent with the design bases.

V. Response to Item (d): Description of Problem Identification and Corrective Action Processes

A. Introduction

1. Purpose

This section responds to the following NRC request for information:

> *"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."*

2. Overview

Department management expects every employee and contractor performing work at, or in support of the Susquehanna Steam Electric Station (SSES) to identify problems, even those which are seemingly minor in nature. Processes are designed to promptly identify problems and to assure resolution through implementation of appropriate corrective actions. The identification of problems is encouraged through the wide variety of identification mechanisms, management communications and behavior, and staff training in problem identification. The Nuclear Department philosophy is that a broad identification of problems provides the optimal defense-in-depth against adverse events or conditions.

This is coupled with a single reporting mechanism under PP&L's corrective actions program. PP&L's corrective action processes implement to 10 CFR 50, Appendix B Criteria XV and XVI. The processes are designed to provide for appropriate application of the following activities:

- formal reporting of the deficient conditions;
- identification of the requirement(s) to which the item/activity does not conform and appropriate reference information to support/substantiate the requirement(s);

- notification to responsible and interested parties;
- control of the deficiency through tagging, segregation, administrative controls or other appropriate means to prevent inadvertent installation or use;
- resolution/disposition by authorized personnel;
- trend analyses;
- a determination of significance that, as a minimum, considers the following:
    - major program breakdown
    - license/regulation violation
    - repetitive nonconforming conditions
    - trends adverse to quality
- for conditions which are significant, the root cause of the deficiency shall be determined and corrective action to preclude recurrence shall be established;
- documentation of the corrective action, and
- provisions for the verification of corrective action implementation.

Within these processes, a deficiency is described as that characteristic of an item, material, component, or document that makes it nonconforming with the original acceptance criteria. The corrective action processes allow for identification and assessment of events or situations which are not necessarily adverse to quality, but which may be precursors to more significant problems or may provide an opportunity to improve performance.

3. Organization

The detailed response to item (d) is divided into three distinct sections, including:

- PP&L Process Framework for Problem Identification and Corrective Action
- Description of Problem Identification and Corrective Action Processes
- Process Oversight

B. PP&L Process Framework for Problem Identification and Corrective Action

The primary basis for the corrective action processes is compliance with the regulatory requirements of 10 CFR 50, Appendix B, Criterion XV and Criterion XVI. The processes also conform to ANSI N45.2.12-1977, Regulatory Guide 1.144, ANSI N18.7-1976 and Regulatory Guide 1.33. The requirements are specified in FSAR Chapter 17, Section 17.2.15, "Nonconforming Materials, Parts or Components" and Section 17.2.16, "Corrective Action"; and the Operational Quality Assurance Manual, most specifically under Operational Policy Statement OPS-5, "Deficiency Control System." Each specific process is procedurally defined.

When a problem, deficiency, or concern is evaluated as a condition adverse to quality, it is handled in accordance with the Condition Report (CR) process as defined within implementing procedures. Procedures require that CRs receive a prompt operability determination and a timely reportability determination, which is consistent with the guidance of Generic Letter 91-18 (Resolution of Degraded and Nonconforming Conditions and Operability), 10 CFR 50.72, and 10 CFR 50.73. When these determinations identify the need to formally report the problem to the NRC, the PP&L processes assure compliance with applicable regulatory requirements.

PP&L's corrective action processes have evolved over time, based upon feedback furnished during internal and external audits, assessments and inspections. As an example of the improvements identified and made, the Engineering Deficiency Report (EDR) process was initiated in 1990 to address the fact that existing corrective action processes, primarily the Nonconformance Report (NCR), were not well suited to design deficiencies and concerns. The EDR process established a formal means to satisfactorily resolve engineering problems previously identified under other corrective action or work management processes.

The EDR process evolved further in 1991, when an Engineering Discrepancy Management work group was established to further enhance Department focus on the corrective action process. This group implemented policies to ensure EDR priority was based on safety significance, and that EDRs would be resolved within one refueling cycle.

In 1993, a team was formed to review the effectiveness of the EDR process, and substantial changes resulted; examples of these enhancements include:

- elevation to a Department-wide process;
- consistency with other Department deficiency management processes;
- lowered threshold for EDR acceptance;
- greater management accountability;
- establishment of time limits on operability, reportability and disposition plan preparation;
- regular status updating to management;
- closure verification; and
- feedback to originators.

As a result of this effort, work began in 1994 to evaluate consolidation of the Department's several deficiency management programs. This culminated in 1995 with the establishment of the current Condition Report process described below.

Both internal and independent, external assessments and inspections have provided positive feedback on the CR process. As with other PP&L's processes, the corrective action process is subject to periodic evaluations targeted at identifying any weaknesses that need attention and correction to assure long-term compliance with regulatory commitments and PP&L management expectations.

C. Description of Problem Identification and Corrective Action Processes

Problems, deficiencies, or concerns reported under PP&L's corrective action processes encompass a broad scope, including operating events, degraded and nonconforming material or parts, inappropriate human actions, employee concerns, procedure violations, apparent discrepancies in configuration control or design, and noncompliance with regulatory commitments or applicable federal and state codes. The various processes all provide for prompt identification and documentation, significance evaluations, and corrective action implementation to provide for resolution.

The Condition Report (CR) is the top level corrective action process that is responsive to 10 CFR 50, Appendix B, Criterion XVI for the identification, evaluation and correction of conditions adverse to quality. As referenced above, in 1995, this evaluation culminated in the combining of four mechanisms (NCR, EDR, SOOR, Audit Finding), all of which were responsive to 10CFR50, Appendix B requirements. With this singular mechanism, Department personnel understand the importance and

significance placed on the process, and Department management now has a singular tool for use in trending conditions adverse to quality.

CRs receive a prompt operability determination (OD), commensurate with the safety significance of the condition. The procedural requirements for these ODs are derived, in part, from guidance in NRC Generic Letter 91-18. PP&L considers the basis for operability as the ability of systems, structures, and components to perform their specified functions as described in the plant design basis, the Technical Specifications, and the FSAR. Reevaluation of operability is a continuous and ongoing process that employs additional data points, including engineering studies and analyses, vendor information, industry experience data, and engineering calculations.

1. Problem Identification

There is a broad range of processes available to report problems, deficiencies, or concerns. These processes ensure that the problem, discrepancy, or concern is reported, evaluated for significance, and is corrected through appropriate corrective actions. Based upon the significance of an individual problem or concern, or an adverse trend in lower significance problems or concerns, the CR process may be used to formally identify the item, evaluate operability and reportability, determine cause, and implement corrective actions necessary to both correct and prevent recurrence of the problem .

The following delineates some other key problem identification mechanisms available for use by Nuclear Department staff and contractors:

- Apparent Configuration Discrepancy (ACD): Discrepancies involving the configuration of the plant are discovered during plant modification reviews, design reviews, plant events or other reviews and investigations. These problems are identified and corrected under the ACD Program. Usually, these are minor document discrepancies, which do not affect the operation or design basis of the plant. However, when these discrepancies represent a condition adverse to quality, a CR is written.

- Design Basis Documentation (DBD) Open Item: As described in Section VI, the Design Basis Documentation (DBD) Project identifies design basis and other discrepancies as open items.

When these discrepancies represent a condition adverse to quality, a CR is written, such that the condition, operability determination, and reportability determination are all addressed under the corrective action program.

- CLB Project "Apparent" FSAR Discrepancies: As the FSAR is reviewed, discrepancies are identified and actions initiated to correct. If the discrepancy constitutes a condition adverse to quality, a CR is written such that the condition, operability determination, and reportability determination are all addressed under the corrective action program. (See Section VI for additional description of the CLB Project.)

- Work Authorization (WA): The WA system is used to identify, control and document work activities associated with plant structures, systems and components. If the condition has potential significant impact on plant equipment the Shift Supervisor is promptly notified to initiate immediate actions to ensure the plant is maintained in a safe condition. A CR is written to control the identification, significance evaluation, and implementation of corrective actions. Other problems associated with plant structures, systems and components are handled via the WA process to ensure proper problem identification, and corrective action implementation.

- In-Process Corrected Error Reports (ICE): The Quality Control (QC) organization identifies errors which occur during physical work on quality components in the plant, which may be corrected during the ongoing work process, and documents these within QC inspection reports. Over 700 of these 'ICE' reports were written in 1996. Trend analysis is performed and adverse trends or a condition adverse to quality are documented via a CR.

- Human Performance Enhancement System (HPES): The HPES Program is a voluntary, non-punitive, independent reporting system which is used to report non-consequential occurrences as well as near misses. In addition, it may also be used to report opportunities for improvement which could help personnel to be more successful in their job performance. Those who use the HPES reporting mechanism to report human performance events are guaranteed full indemnification from reprisal. Individuals document these type of events and identify the apparent causal factors via a simple reporting form.

These reports are reviewed by an independent assessment group, which issues a CR if the problem is considered a condition adverse to quality.

- Employee Concerns Program (ECP): The purpose of the ECP is to ensure that individuals can raise concerns and have them addressed without concern for retaliation. It provides an effective safety net for nuclear safety issues that do not get addressed via other methods. The key elements of the program are the availability of confidentiality, and feedback to the individual raising the concern. The ECP has the active involvement of the Vice President - Nuclear Operations.

  There are clear expectations in the Nuclear Department that individuals are encouraged to report issues involving nuclear safety to their supervisor. Individuals who choose not to report an issue to their supervision, or who believe that a previously reported item has not been adequately addressed, are encouraged to contact an Employee Concerns Representative or the NRC. The key to our program is the identification of the concern so that it can be fully addressed and resolved.

  The ECP provides an alternate and confidential method for individuals to identify concerns. If a nuclear safety concern is identified to an Employee Concerns Representative, a CR will be generated. The ECP then follows the disposition of the CR and ensures that it is resolved. If needed, necessary resources, including non-PP&L professional experts, in cases of difference of professional opinions, are used to investigate concerns. Employee concerns are closed only when all implementing corrective actions are completed and the concerned individual agrees to close them. If not satisfied with the resolution, the individual has the option of submitting the concern to the NRC.

- Industry Events Review Program (IERP): PP&L implements a program for the evaluation of industry operating experience information for impacts on SSES. The program is defined procedurally, and utilizes information received via NRC Information Notices, Bulletins, Generic Letters, and NUREGs; applicable vendor information including 10 CFR 21 notifications; GE Service Information Letters (SILs, TILs), and Service Advisory Letters (SALs); and INPO Significant Event Evaluation, Information Network items, and Good Practices.

Following a preliminary screening by Nuclear Licensing, the information is forwarded to the appropriate Nuclear Department functional unit (System Engineering, Operations, Maintenance, Procurement, etc.) for an in-depth evaluation and initiation of appropriate corrective actions. Those that have a real or potential impact on SSES are identified on CRs. This facilitates the conduct of operability and reportability evaluations, as well as the development and implementation of corrective action plans necessary to maintain/revise the design/operation of SSES.

- Many other controlled and uncontrolled processes provide for identification, evaluation, and resolution of problems which provide potential inputs to the CR process. These include:

  -- Maintenance Rule
  -- Leakage Rate Testing Program
  -- Surveillance Program
  -- Scram Open Items
  -- MOV Program
  -- Improved Technical Specifications (ITS) Project
  -- Employee ALARA Concern (EAC)
  -- Outage Task Improvement Recommendations ("Green Cards")
  -- Opportunity For Improvement

- NAS Oversight Activities: The Nuclear Department has in place a wide variety of self-assessment activities to identify and correct problems. These include:

  -- A program of QA Audits to verify compliance with regulatory commitments. This program is conducted in accordance with the requirements of 10 CFR 50, Appendix B, Criterion 18, and ANSI N45.2.12-1977. The scope of these audits includes all programs and processes established under the Operational Quality Assurance program, as well as audits specified in Technical Specifications for the Susquehanna Review Committee. As such, audits are scoped to evaluate the maintenance of the design bases and conformance of the plant configuration, maintenance, and operations with this bases. The results of these audits are reported to appropriate levels of management and provide for the identification and resolution of problems through the use of a CR.

-- A program of both scheduled and unscheduled quality surveillances to observe ongoing activities and allow for timely evaluation of these activities. The scope of scheduled surveillances is defined by a matrix of areas or categories which are reviewed on a 2-year cycle. Unscheduled surveillances are performed as concerns occur or at the request of managers. The results of these surveillances are reported to appropriate management and utilize the CR and HPES for problem identification and resolution.

-- A program of independent assessments and self-assessments conducted throughout the Department at the direction of responsible managers. The scope of these range from pre-job reviews to event reviews and structured process analyses. The Department compiles an integrated assessment plan each year to describe planned self-assessment activities. The results of these assessments are documented, evaluated for significance, and resolved by appropriate Department management. Problems identified within these assessments that are deemed to be conditions adverse to quality are entered into the CR process.

-- Independent Safety Evaluation Services (ISES) performs the function of the Independent Safety Engineering Group (ISEG) originally defined in NUREG-0737, and later required by the SSES Technical Specifications. ISES examines the SSES operating characteristics, NRC issues, industry advisories, Licensee Event Reports (LER), and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety. Specific activities include:

(1) Conducting safety assessments of Department activities, including investigations of operational incidents;
(2) Evaluating technical adequacy and clarity of those procedures important to the safe operation of SSES;
(3) Evaluating plant performance and operations from a safety/environmental conformance perspective; and
(4) Reviewing violations, deviations, and reportable events at SSES that require written reporting to the NRC.

The results of these examinations are reported to appropriate management for the initiation of appropriate actions. When conditions adverse to quality are noted, the CR process is utilized to identify, evaluate and correct the problems.

-- Quality Control Services (QCS) plans and performs inspections during plant maintenance and modification activities. The acceptance criteria for these inspection activities are obtained for engineering documents (DCP, ECO, specifications, drawings, work plans) that are traceable to the design bases. During conduct of the inspections, plant walkdowns and ancillary review of record packages, the QCS staff provides for another level of assurance of the conformance of plant configuration with the design bases. The resolution of problems noted in these inspections are handled via the CR or ICE processes, as previously described.

Problems identified via these mechanisms are periodically reviewed. If specific problems, or a trend of lower-level precursors, constitute a deficient condition, a CR is written.

2. Condition Report Process

The Condition Report process provides policy and direction to ensure that conditions adverse to quality are identified and resolved in a manner that: ensures safe and reliable operation of SSES, complies with regulatory requirements, fosters an environment which encourages the identification and timely resolution of safety and quality concerns. Integral to this process is timely determination of operability and reportability pursuant to regulatory requirements associated with identified conditions.

a) Event Reporting and Initial Actions

When conditions adverse to quality are identified, they are documented on a CR. Operations shift supervision is immediately notified when there is known impact to plant operation. In addition, the following actions are initiated.

- Immediate investigation by Shift Supervision for significant operating occurrences;

- Operability Determination in accordance with procedures; and

- Reportability Determination per plant procedures to evaluate immediate and prompt NRC notification requirements.

b) *Initial Investigation*

CRs receive an independent investigation by Nuclear Assessment Services usually the next business day following the first daily (7:30 AM) plant meeting after the event or condition is identified. This investigation (Significance Review) includes, but is not limited to, the following activities:

- Classification of the Event/condition

  This event classification includes separate categories for the following engineering deficiencies:

  - Design
  - Configuration Control
  - Current Licensing Basis (CLB)
  - Design Basis

  Classification into these categories, particularly the last two, helps ensure that specific conditions, or trends, involving the design basis of the plant receive the necessary review, analysis, and oversight.

- Significance of the Event/Condition

  Actual and potential consequences of the condition, generic implications, and previous operating experience and trends are assessed. Conditions where the plant is found outside of design basis or in an unanalyzed condition are assigned a Level 1 (most significant) rating, which requires a formal root cause analysis, safety assessment, and actions to correct condition and prevent recurrence.

c) *Corrective Action Team (CAT) Review*

The results of these independent CR significance reviews are usually presented each work day to a Corrective Action Team (CAT), which normally includes the Plant Manager-Susquehanna, Manager - Operations, Manager-Nuclear Engineering, Manager-Maintenance, Manager-Outages, Manager-Plant Services, Manager-Nuclear Assessment Services, and the three engineering Functional Unit Managers. Representatives attend for these Managers when they are unavailable. As a result, conditions adverse to quality receive prompt and high level management involvement in initial event review and corrective action planning.

The intent of the CAT Meetings is to ensure timely and thorough management involvement in the corrective action process. They provide direction on work group assignment, the investigations and pertinent issues, discuss generic issues and trends, and ensure that immediate corrective actions and operability determinations are adequate.

d) *Evaluation and Action Plan*

Causes and causal factors are identified for CRs. For conditions which are not significant conditions adverse to quality, the causes are determined and documented during an initial independent investigation of the reported problem, and the condition must be corrected. For significant conditions adverse to quality, a more intensive formal cause determination and/or root cause analysis is conducted within 30 days (20 days for CRs requiring a License Event report (LER)). These evaluations require a documented investigation, safety assessment, discussion of generic implications, review of past internal and industry operating experience, identification of root cause(s), and actions to prevent recurrence and correct the condition. Interim corrective actions are considered in the corrective action plan.

Management has discretion to direct these same requirements on non-significant conditions adverse to quality.

e) *Action Tracking*

Actions to correct the condition and prevent recurrence are documented on a separate "Action Required" document and issued to the responsible work group for implementation. When the action is completed, it is documented and verified by the responsible individual, and then reviewed and approved by the responsible manager. The action tracking data and status is maintained in a computer database and the information is available on-line to all work groups for their review and use in ensuring timely closure of the actions. Periodic status reports are issued to all responsible management and discussed weekly at CAT Meetings.

f) *Implementation of Corrective Actions*

Corrective action implementation is intended to be completed within a reasonable time depending on the significance and nature of the problem and specific actions. Management expectations are that corrective actions not requiring a plant modification or unit outage will be completed within 6 months from the finalization of the corrective action plans. The action tracking process discussed previously is utilized to facilitate timely implementation of corrective action commitments.

If corrective actions are not completed within one fuel cycle, the Lead Functional Unit Manager must reevaluate the Operability Determination, document a written justification for the schedule extension, and obtain management approval.

g) *Deficiency Closure*

After corrective actions are completed, the CR is closed by the Operating Experience Services organization following verification by the responsible managers that all actions have been completed. The CR documents are retained in Nuclear Department records for future retrieval.

h) *Trend Analysis*

The process provides for the performance of various types of trend analysis by the Operating Experience Services staff in conducting their independent significance reviews, the CAT in evaluating problems for generic/repeat implications, functional management in development and implementation of corrective action plans, and the independent management and oversight staffs in evaluating the effectiveness of the process. Trending data is available by event codes and cause codes.

D. Process Oversight

1. Management Review

Resolutions (root cause analysis, safety assessment, actions to correct condition and prevent recurrence) of Level 1 Condition Reports and NRC reportable CRs require independent review and/or approval by the following:

- Lead Functional Unit Manager (Approval)
- Affected Functional Unit Managers (Approval)
- Supervisor, Operating Experience Services (Approval)
- PORC (Approval)
- SRC (Review Only)

The intent of these independent reviews and approvals is to assure that the appropriate considerations were addressed in problem evaluation and corrective action development including generic implications considerations, impact on design basis, operability, and reportability.

Level 2 Condition Reports require the approval of the lead and affected functional unit managers, and Levels 3 and 4 CRs are reviewed by CAT.

2. Review Committees

PP&L has established several independent review committees that are responsible to conduct various reviews including those of proposed changes in procedures, the facility as described in the SAR, the Technical Specifications including the associated written safety evaluations required by 10 CFR 50.59 and assure compliance with design, licensing, and operating bases including the positive closure of problems and reportable events. These independent committees function as advisory panels to either the Senior Vice President - Nuclear, Plant Manager - SSES, or the Manager - Nuclear Engineering. The specific committees and associated responsibilities as related to the corrective action processes are outlined below:

a) Susquehanna Review Committee (SRC)

The SRC is a review, audit, and advisory group, composed of at least five key Nuclear Department managers as well as members external to PP&L, whose function is to verify independently that the SSES is being operated and maintained in accordance with safety-related, ALARA, and environmental requirements. The SRC performs the offsite independent reviews mandated by ANSI N18.7, FSAR Chapter 13.4.2, and SSES Technical Specifications. The SRC reports to the Senior Vice President - Nuclear. Specific SRC review responsibilities related to the corrective action processes include the following

- Violations of codes, regulations, orders, Technical Specifications or of internal procedures or instructions having nuclear safety significance;

- Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

- Reportable Events;

- Recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

- Reports and meeting minutes of the PORC.

Under its audit responsibilities, the SRC mandates that periodic audits be performed of the results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety; the performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50; and any other area of unit operation considered appropriate by the SRC or the Senior Vice President - Nuclear. These SRC mandated audits are normally performed by the Nuclear Assessment Services staff as an integral part of the annually scheduled audit program with the results formally reported to and reviewed by the SRC.

b) Plant Operations Review Committee (PORC)

The PORC is responsible for functioning in an advisory role to the Plant Manager on matters of nuclear safety. The PORC performs the onsite independent reviews mandated by ANSI N18.7, FSAR Chapter 13.4.1, and SSES Technical Specifications. Specific PORC review responsibilities related to the corrective action processes include the following:

- Investigation of violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Senior Vice President - Nuclear and to the Chairman of the SRC;

- Review of events requiring notification to the NRC under 10 CFR 50.73.

- Review of unit operations to detect potential nuclear safety hazards.

c) *Engineering Review Committee (ERC)*

Although not required by the Technical Specifications, the ERC was established by PP&L in an oversight and advisory role to the Manager - Nuclear Engineering primarily on matters of nuclear and environmental safety, as well as matters relating to the quality of engineering activities undertaken in support of SSES. Specific ERC oversight and review responsibilities regarding the Corrective Action Program include the following:

- Perform special reviews, investigations, and analyses on any aspect of SSES engineering operations as assigned by the ERC Chairman to assess the quality of those operations and their effect on the safety of SSES and its environment;

- Review assessments of engineering and design errors including root cause assessments of deficiencies found in engineering products and processes;

- Review of items uncovered in ERC reviews, or in activities outside of ERC, which may be considered a potential threat to nuclear safety, or which may be indicative of a broad based degradation of engineering quality; and

- Review the general quality level of the SSES design modification and reload design programs. Examples of specific types of documents reviewed include: Design Standards and Design Guides; specifications;

engineering products such as Safety Evaluations, Technical Safety Assessments, drawings and other engineering document processes; CRs related to engineering activities; and NAS Audit Reports related to engineering activities.

d) *Nuclear Oversight Committee*

The Nuclear Oversight Committee (NOC) is responsible to the Board of Directors for identifying those matters involving the nuclear function which warrant the attention of the full Board, for providing recommendations concerning the future direction of the Company relating to nuclear operation, and for communicating the Committee's judgments and perceptions of management performance in this area. Committee deliberations focus on management systems and processes which address and provide insight to:

- Issues critical to the long-term safe, reliable and economical operation of the generating units.
- Major decisions and significant management issues for the future.
- The effectiveness of management.
- Human resources.

E. Communications to the NRC

Discrepancies which involve the design of the plant are identified through one of the methods discussed above. If the design basis of the plant could potentially be affected, a CR is written. CRs involving design basis discrepancies are immediately reviewed for reporting requirements by the Operations Shift Supervisor and/or Supervisor, Operating Experience Services. If the plant is discovered in a condition outside of its design basis, the condition is reported to the NRC per 10 CFR 50.72(b)(l)(ii) and 10 CFR 50.73(a)(2)(ii). Guidance from NUREG 1022 is used in making these reportability determinations, as well as joint evaluations by Operations, Licensing, Engineering, and Operating Experience Services. If

not reportable under 10 CFR 50.72 or 10 CFR 50.73, 10 CFR 21 is also reviewed if a potential defect or noncompliance exists.

There are both required and voluntary communications with the NRC on an ongoing basis. The portion of required reporting which have some relationship to maintaining the design basis of the plant includes the following:

- 10 CFR 50.71(e)(4), Annual FSAR Update Report
- 10 CFR 50.59(b)(2), Annual Report of Changes, Tests, and Experiments
- 10 CFR 50.9(b), Notification of Information with Significant Implications for Public Health or Safety
- 10 CFR 50.54, Reporting Changes to License Conditions, including Emergency Plan, Security Plan, and Operational QA Program
- Operating Licenses NPF-14 and NPF-22
- SSES Technical Specifications

In addition to the regulatory requirements, other communications with the NRC ensure that they are kept informed of emerging issues and problems. The onsite NRC Resident Inspectors are provided problem identification and supporting documentation upon the identification of the problem and periodically as PP&L works through the processes. The Plant Manager - SSES meets frequently (usually once a week) with the Senior NRC Resident Inspector for an open and informal discussion of current plant issues and problems. Nuclear Licensing, the Manager of Nuclear Engineering, the Vice President - Nuclear Operations, and the Senior Vice President - Nuclear all maintain open communications with both NRC Regional staff and NRR staff.

F. Conclusion

PP&L has established mechanisms for problem identification and corrective action related to the implementation of the design and configuration controls (i.e., "translation" processes) described in Sections II through IV. Effective implementation of our problem identification and corrective action processes allows us to correct inconsistencies between the design bases and the plant, and to prevent recurrence of these problems with feedback to the design and configuration control processes. This ensures that processes evolve through lessons learned from operating experiences.

VI. Response to Item (e): Overall Effectiveness of Processes and Programs

A. Introduction

1. Purpose

This section addresses item (e), which requests a description of:

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

2. Overview

Our response to item (e) draws primarily upon the information provided in our responses to items (a) through (d), as described in Sections II through V. From this information, this section derives the key points listed below regarding the effectiveness of our current processes.

- Original design, construction, procurement, licensing and start-up of the Susquehanna Steam Electric Station (SSES) were conducted under quality assurance controls and practices, which established a solid foundation for today's control systems. PP&L took early ownership of the design and configuration of SSES through extensive engineering turnover and initial testing activities.

- Our current integrated set of processes for controlling changes to, and deficiencies in, the design and configuration of the plant was also developed based upon the QA program framework. These processes provide mechanisms for translating design bases changes into plant procedures, configuration and performance. These processes have evolved over time and reflect experience gained throughout our own operating history, as well as from industry events. The knowledge-base of our personnel in process implementation has also developed from the baseline described above. The key management systems, values, and quality principles underlying the operation and maintenance of SSES provide the tools necessary to implement the processes, and to improve the processes.

- Quality assurance principles are embedded in the manner in which we perform work, and provide opportunities for feedback on the effectiveness of our translation of design bases into plant procedures, configuration and performance. We have found problems in the past, through oversight and assessment (internal and external) activities, as well as during the course of major projects (e.g., Power Uprate Program, Improved Technical Specification Project, Design Basis Documentation Project, etc.), and have taken, and will continue to take, actions to correct and prevent recurrence of such problems. Where appropriate, actions have included, and will include process enhancements. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that design bases have been effectively translated into plant procedures, configuration and performance. Finally, as a further means of identifying opportunities to enhance processes, we initiated a three-phased Current Licensing Basis (CLB) Project in early 1996. Phase III of this project is anticipated to begin in mid-1997 and will be focused on many of the issues identified in the subject NRC request for information.

3. Organization

In support of these three points, the response to item (e) includes the following subsections:

- Design Baseline
- Integrated Set of Design, Configuration and Deficiency Controls
- Feedback on Process Effectiveness

B. Design Baseline

As described in Section II, the original plant design "baseline" was established in a rigorous and systematic manner. Construction and engineering turnovers and independent design reviews provided reasonable assurance regarding the adequacy of the design information

describing the plant and the consistency between the physical plant and this design information. This information was developed, verified, controlled, and utilized under quality assurance practices that were responsive to 10 CFR 50, Appendix B, and provided a foundation from which PP&L controlled the design and configuration of SSES.

Section II further describes that the construction turnover walkdowns and design reviews verified conformance of the plant configuration with design output documentation. The engineering turnover process, which involved independent design reviews, provided reasonable assurance that the design met technical standards and that design bases were sufficiently complete so as to support plant operation, maintenance, testing and modification activities. The Initial Test Program provided design verification by confirming that design parameters were within required ranges, and consistent with data submitted to the NRC. Therefore, we have reasonable assurance that the original design was consistent with technical standards, the physical plant matched the design basis information, and the design basis information turned over to PP&L was adequate to enable PP&L to safely operate, maintain, and modify SSES within the bounds of our operating license.

PP&L established early ownership of the design and configuration of SSES through extensive turnover and start-up testing activities. A quality assurance (QA) program was developed for the design and construction phases of the SSES project, which included oversight of the implementation of QA requirements. Furthermore, the significant involvement of PP&L's Integrated Start-Up Group throughout the start-up process established an understanding of the importance of consistent translation of design bases into plant procedures and the plant configuration. PP&L's definition and execution of an extensive engineering turnover process provided additional assurance that the design bases information necessary to maintain the plant and its design margins was available.

As described in Section III, PP&L's early involvement in the development of plant procedures, in parallel with the start-up testing, helped to assure that the plant design, as translated into start-up testing, was also translated into plant procedures. Section IV points out that PP&L's involvement with turnover walkdowns and initial testing established an understanding of the importance of consistent translation of the design bases into the physical plant configuration, as well as an understanding of the value of testing as a means of confirming consistency between operating system performance and design bases.

C. Integrated Set of Design, Configuration and Deficiency Controls

From this baseline, PP&L has maintained ownership of the design, operation and maintenance of the plant using qualified and experienced personnel, and appropriate procedures. The knowledge-base of our personnel has continued to grow, in turn allowing PP&L to retain ownership of design, rather than relying heavily upon contractors for design work. Additionally, the QA framework was expanded upon by a set of Supplemental Procedures to define and facilitate the transition between construction and operational phases of SSES, and ultimately by the development of the PP&L Operational QA Program.

1. Process Framework

As described in the preceding sections, we have established an integrated system of procedures controlling the design and configuration of SSES, including corrective action processes, which have evolved over time. The process descriptions provided in Sections II through V illustrate how we ensure consistency between analogous processes, and provide appropriate linkages between procedures.

Within the framework of the quality assurance program, our processes have continued to evolve. As described in Section II, prior to start-up, PP&L recognized the need to make the transition between the design processes utilized during the design and construction phases to a set of controls more appropriate for changes made during the operational phase. Section II of the attachment describes PP&L's current integrated set of design and configuration control processes. These processes include mechanisms for considering the design bases during design, transforming a design change into information describing the configuration of the plant, and then finally for implementing design changes into the actual physical configuration of the plant.

Section III addresses the questions of how design bases were originally translated into plant procedures, and how design changes are currently translated into plant procedures. Processes have been established to assure that as the plant is modified, procedures are reviewed by knowledgeable personnel and are appropriately updated to translate the design change. In addition to periodic reviews to determine whether changes are necessary, operating, maintenance and testing procedures are reviewed, and revised as

necessary, following events in which the procedure contributed to the cause of the event or was inadequate in mitigating the effects of the event.

Section IV addresses the processes used to translate the design changes into the plant configuration. As described in Section II, the configuration management requirements are an integral part of the plant modification program. The modification process itself is designed to assure that the design bases are identified, are properly converted into the design, are reflected in design output documents and then translated into the actual physical plant configuration. For example, the process used to install and close-out Design Change Packages (DCPs) includes development of a Plant Modification Package by engineering and plant personnel to identify all work documents, procedure changes and testing requirements necessary to implement the modification.

Section V addresses our mechanisms for problem identification and corrective action. These are the processes we use to identify anomalies in the design and configuration controls (i.e., "translation" processes) described in Sections II through IV. Implementation of our problem identification and corrective action processes allows us to correct inconsistencies between the design bases and the plant configuration, performance and procedures, and to prevent recurrence of these problems through feedback to the design and configuration control processes. This assures that processes evolve through lessons learned from operating experiences.

The process descriptions provided within this response illustrate our attention to design change control activities. Our mechanisms for problem identification and continuous improvement have been directed at ensuring that the generic implications of deficiencies have been addressed, and processes enhanced accordingly. Therefore, our processes reflect lessons learned from past deficiencies and identified weaknesses. One example of our process evolution in the area of corrective action is presented in Section V, which notes that the Engineering Deficiency Report (EDR) was developed in recognition of the need to provide a corrective action mechanism focused on engineering deficiencies. In the last two years, a consolidation of corrective action processes was performed, and the EDR and other mechanisms evolved into the Condition Report (CR) process, which is currently in use. The CR simplifies the corrective action process for the user, while

incorporating the "best-of-the-best" from each of its predecessor programs.

2. Process Implementation

PP&L's Nuclear Department management has established and communicated a goal of achieving excellence in the operation, maintenance and support of SSES. PP&L's approach to nuclear operation is predicated on the acknowledgment of its obligation to provide for safe and efficient operation, and the intent to place the safety of the public and employees above the economic benefits to the company and its customers. This is evidenced, in part, by PP&L's "ownership" of the design from the design, construction and start-up phases of SSES, as described in Section II.

We have set forth quality assurance principles and practices and established tools that provide personnel a means to achieve management expectations. Key in this regard is access to accurate design basis and configuration information. Examples of some of these information tools include ready access to PP&L design basis calculations, the developing Nuclear Information Management System (NIMS), and electronic access to a variety of licensing and design basis information sources. (See Section II for details).

We maintain a work environment that attracts and retains capable people. As described in Sections II, III and IV, many of our people have been involved with SSES from the early design and construction phases of the plant and have continued through to our current operational phase. This institutional knowledge base, which has evolved over our operating history, provides added assurance and awareness of the importance of maintaining consistency between the plant design basis and the plant configuration.

Our workforce is highly qualified and experienced. As described in Section II, we have established supplemental training and certification programs. For example, for design and system engineers, we have developed a continuing engineering training program which incorporates lessons learned from both industry and within PP&L. We train our personnel on design control and configuration control, as well as licensing basis and design basis issues.

As described in Section II, Department management is involved in the design and configuration control, and corrective action processes. Examples include the involvement of management review committees, such as the Plant Operations Review Committee (PORC) and the Engineering Review Committee (ERC), in the review of plant modifications and procedure changes. Further, as described in Section V, management participates daily in reviewing CRs, on the Corrective Action Team (CAT).

Therefore, we have confidence in the key management systems and values that underlie our processes and support implementation of the processes. These provide the means to assure effective implementation of the processes and programs.

D. Feedback on Process Effectiveness

As described in Section V, management has provided personnel with mechanisms to identify problems and opportunities to enhance processes. The wide variety of mechanisms for problem identification, and emphasis by Department management on quality assurance principles, provide confidence that design-related deficiencies are captured, and that opportunities for process enhancements are identified.

Process improvement results from quality assurance oversight and assessment (internal and external) activities, as well as through feedback from process implementation. Over the operating history of SSES, there have been a number of indicators of the effectiveness of our processes and programs for maintaining the consistency of the plant configuration with design bases. One set of indicators includes the results of quality assurance audits and inspections of our design and configuration control processes.

Also, embedded within our execution of major plant modifications and projects are the quality assurance principles which assure that design discrepancies are identified and corrected, and that process enhancements necessary to prevent recurrence are identified. Examples of these projects include the Power Uprate Program, the Improved Technical Specifications project, and the Design Basis Documentation project. These projects have required extensive review, and thus have provided significant indications of the effectiveness of our design and configuration control processes. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added

confidence that design bases have been effectively translated. Successful translation of design bases into plant configuration is also supported by the fact that plant performance and response to transients have been consistent with design bases analyses.

Our ongoing Current Licensing Basis (CLB) Project will further evaluate the design bases issues raised in your request for information. This initiative was launched in early 1996 in response to industry events, prior to the issuance of the referenced NRC letter. PP&L has already identified discrepancies and improvement opportunities as a result of ongoing scoping reviews of the FSAR, and expects that more will be found. None of the deficiencies identified to date have been safety significant.

1. Oversight and Assessment Activities

Audits and assessments (internal and external) have provided adequate confidence in the effectiveness of our processes and programs. Note that in some cases, these reviews have identified various weaknesses related to control of the design bases or consistency between the design bases and the physical plant. Upon identification, these items are entered into the corrective action process where they are evaluated for safety significance, operability, and reportability and where appropriate, corrective actions are implemented.

a) Audits

In response to 10 CFR 50, Appendix B Criterion XVIII, "Audits," PP&L, via commitments delineated first in Appendix D of the Preliminary Safety Analysis Report (PSAR) and currently in the Final Safety Analysis Report (FSAR) subchapters 13.4.2.9, 13.4.3, and 17.2.18, established and implemented a comprehensive system of planned and periodic audits to verify compliance with PP&L quality assurance policies, principles, and practices, approved operating procedures, license provisions, and administrative controls as defined within the Operational QA Program. Periodic audits of programs for design control, configuration management, procedure control, test control, and plant operations and maintenance have been conducted since the start of initial design and construction.

These audits, through a review of objective evidence, observations, walkdowns and interviews, evaluate the adequacy and effectiveness of the quality assurance practices, procedures and processes in ensuring that design bases are consistently and adequately translated into the physical plant, as well as operating and maintenance procedures. The results of these audits also demonstrate that reasonable assurance of long-term compliance with original design bases is furnished through the effective implementation of the established quality assurance practices and programs. As was discussed in Section V, when audits identify problems in either the adequacy or implementation of the established quality assurance practices and programs, responsible Department management initiates an evaluation of the problem and implements appropriate corrective actions to both correct and prevent recurrence. PP&L's formal audit program has evolved over the life of SSES, and today is supplemented by a broad-based assessment program which includes both independent assessments and self-assessments throughout the Nuclear Department.

Historical audit data indicates that the established programs and quality assurance practices have been generally effective in assuring consistency of the physical plant, operating and maintenance procedures, and design bases. Additionally, the results of PP&L assessments and third party assessments, including NRC inspections, have generally substantiated both the adequacy, and effectiveness of, the established quality assurance practices and programs. When these internal and external assessments identified program or implementation weaknesses, PP&L management initiated appropriate corrective action directed at both resolving the issues, and precluding recurrence of similar conditions.

PP&L's audit and QA oversight activities during the initial design, procurement, construction, turnover of systems, pre-operational testing, and start-up testing included evaluations of the adequacy of the translation of design bases into the plant configuration, operating, maintenance, and testing procedures. The audits conducted during this phase of the SSES project were directed toward Bechtel, GE, and PP&L programs and quality assurance practices, and included corporate office and onsite activities. QA

provided an independent inspection function during the conduct of pre-operational testing activities. QA provided an independent review of all start-up test procedures and results, as well as all plant procedures. The results of these independent activities revealed that the established processes were adequately implemented during performance of the initial design, procurement, construction, turn-over and testing activities. Problems noted during these oversight activities were handled via the corrective action processes.

Subsequent to PP&L's acceptance of turned-over structures, systems, and components, the QA audits of the design, procurement and configuration control processes included evaluations of the implementation of quality assurance practices and procedural controls established to provide reasonable assurance of long-term conformance with design bases during plant operations, maintenance, and modification activities. These audits continue to be used today to determine whether or not the established quality assurance practices and procedures are adequate in assuring proper translation of design into the appropriate documents and are being effectively implemented. A review of audits of these processes, and the products resulting from these processes, revealed the establishment of adequate programs that, for the most part, have been effectively implemented by knowledgeable, well-trained personnel. As is true for the identification of problems, where the audits did note noncompliance with established requirements, the conditions were addressed via the corrective action program.

b) Recent NRC Inspections

As discussed in Section IV, NRC SSFI #90-200 was performed by the NRC in August 1990 on the Electrical Distribution System to determine if the electrical distribution system would be capable of performing its intended safety function as designed, installed and configured. The plant electrical distribution system is perhaps the most widely distributed system in the plant, touching virtually every portion of the plant. The inspection team concluded that generally the SSES

electrical distribution system would be capable of performing its intended safety functions. With the exception of the 14 specific findings identified in the report, the batteries, emergency diesel generators, switchgear, and other components within the Electrical Distribution System were found to be adequately sized and configured. Separation between redundant trains or divisions was found to have been adequately maintained, and an adequate design basis existed and was being upgraded and maintained for the SSES.

The last Systematic Assessment of Licensee Performance (SALP) conducted of SSES covered the period February 27, 1994 to August 5, 1995 (Report Nos. 50-387/95-99 and 50-388/95-99). The NRC team observed a "superior level of performance" at SSES, and assigned Category 1 ratings to all functional performance areas. Strong performance in the maintenance, engineering and plant support areas continued during the SALP period, and improved performance was noted in the operations area. The report specifically stated that "there was excellent communication among departments, effective coordination of activities, and strong evidence of teamwork resolving safety issues." Also, station-wide self-assessments were found to be critical and responsive to improving plant processes. In addition, the recent changes in the handling of station deficiencies through the Condition Report process was perceived to be a positive initiative to improve corrective action activities.

Numerous other NRC inspections related to design and configuration control have been conducted. A review of 1995 and 1996 inspections reveals that in general, current PP&L processes for design and configuration control of SSES have been effective. Although a number of recent findings in a 1996 engineering inspection have focused attention on the need to enhance the use of the licensing basis in the corrective action process, and to reemphasize its importance to our personnel, PP&L self-identified the issues in question, and had previously launched proactive efforts to evaluate similar issues (see discussion of CLB Project below).

c) *Independent Assessment of Design and Configuration Control Processes*

In support of this response, PP&L contracted S. Levy Incorporated to perform a management assessment of the plant design and configuration control processes for SSES. The assessment included a review of Operational Policy Statements and plant change procedures and personnel interviews. The reload fuel change process was reviewed in detail to assure that the plant change processes were comprehensive. In addition, the assessment included a review of PP&L's information management systems, and a review of the Phase I report of the CLB Project, described later in this discussion.

The reviewers concluded that PP&L has developed and implemented reliable, comprehensive procedures, and that information systems facilitated implementation. Some relatively minor areas of the process which have a potential to create problems were identified, but the assessors stated that comprehensive self-assessment is being performed and appropriate actions are being taken to correct the identified problem areas. Finally, the management assessment team noted that it did not identify any new problems that PP&L self-assessments had not already identified, and that appropriate actions are being taken.

2. Major Projects

PP&L has also undertaken major projects which provided unique opportunities for feedback on the effectiveness of design and configuration control processes used to translate design bases. Extensive reviews were conducted as part of our Power Uprate Program and our Improved Technical Specifications project. These projects, along with the development of 18 Design Basis Documents, are examples which provide added confidence that processes have maintained consistency between our design bases and the plant configuration. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that design bases have been translated into plant procedures, configuration and performance.

a) *Power Uprate Program*

The Power Uprate Program (PUP) was initiated in 1991 to increase the power output of both SSES units by approximately 5%. The objective was to apply the design margin that existed between the originally licensed thermal power and the power level for which the plant equipment had been designed and sized. Completion of the PUP required amending the operating license and implementing numerous changes to the plant configuration, design basis information and plant procedures. In 1994, SSES Unit 2 achieved a successful power uprate of about 5%. Unit 1 was similarly uprated in 1995.

A critical part of the PUP was the review conducted to ensure the capability of plant systems and structures affected by the uprated conditions. The PUP reviews were documented in calculations. NEDO-31897, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," provided the basis for the review. An SSES-specific review of balance-of-plant (BOP) systems was also conducted by PP&L.

The task of reviewing plant systems and structures was performed by General Electric and a dedicated team of PP&L engineers. The review of some specific components and/or systems was subcontracted to outside engineering firms when it was determined that a specific knowledge set was needed. An example of this practice was the contracting of the Class 1 piping analysis to Bechtel, the original SSES architect engineer.

A separate calculation was established for each system, structure or topical review. Each review document contained an assessment of the system's or the structure's ability to perform its design function at uprated conditions. The reviews also contained any requirements or recommendations the reviewer considered necessary for successful operation of the system or structure. These documents received a broad review within PP&L's Nuclear Department. The document reviews conducted by the Operations Group and the Plant Systems Engineering

Group focused on system performance and system configuration. Comments on the review documents were addressed and the system review documentation was revised as appropriate.

In addition to system performance assessments, each system review contained recommendations for Technical Specification and FSAR changes. As the recommendations were dispositioned, the design and licensing documents were revised to document the resolution to the recommendation.

The system, structure and topical reviews conducted by PP&L and General Electric formed the technical basis for the SSES Power Uprate Licensing Topical Report. As a measure of the extensive nature of this review, the following topics were evaluated in the report:

- Reactor Core and Fuel Performance
- Reactor Coolant System and Connected Systems
- Engineered Safety Features
- Instrumentation and Control
- Electrical Power and Auxiliary Systems
- Power Conversion Systems
- Radwaste Systems and Radiation Sources
- Reactor Safety Performance Features
- High and Moderate Energy Line Breaks
- Environmental Qualification
- Individual Plant Evaluation
- Emergency Operating Procedures
- Start-Up Testing
- Environmental Assessment

Following the initial draft of the Licensing Topical Report, a PORC Subcommittee was established to review the SSES approach to the power uprate. The PORC Subcommittee review consisted of a cross-disciplinary review of the modification to ensure that the proper PP&L reviews had been performed. Following the PORC Subcommittee review, the full PORC reviewed the PUP, and then a review was performed by the PP&L SRC (first a subcommittee review followed by presentation to the full SRC).

The development of the Power Uprate Licensing Topical Report took approximately two years. It took approximately two more years to implement the uprate on the first SSES unit. During this time modifications were being implemented at SSES. To ensure that these modifications, as well as those performed since the initial licensing of the plant, were compatible with the proposed uprate, a review was conducted. The review was documented in a calculation and issued concurrently with the submittal of the Power Uprate Licensing Topical Report. Administrative controls which conservatively required consideration of uprated plant conditions for plant modifications were also instituted at this time.

Prior to the implementation of the power uprate on the first SSES unit, an operational readiness review was conducted by the PP&L QA Assessment Group. The review was conducted by members of the Assessment Group and independent outside contractors. PP&L arranged an independent assessment consisting of a technical and programmatic review of activities in support of the power uprate. At the time of the assessment, the majority of the engineering analyses had been completed. The assessment team conducted the assessment through a review of documents prepared to support PUP efforts, personnel interviews, observations, and some system walkdowns. The assessment team found no hardware or operational issues that would prevent SSES from achieving the proposed uprated conditions. The review resulted in an Assessment Report with observations and comments. Observations and comments were resolved prior to the implementation of the uprate.

The engineering evaluations for power uprate presented an opportunity to review original design bases and design in a more challenging environment. The knowledge level and understanding of the current engineering staff had matured and in many cases the analysis methods are much more advanced. PUP reviews resulted in seventeen EDRs. The majority of these EDRs involved fuel pool cooling, the spray pond and those systems associated with long-term decay heat removal and compartment temperatures following a loss of coolant accident. The seventeen EDRs were dispositioned and, in several isolated cases, the design bases documentation was supplemented.

Following the PUP implementation refueling and inspection outage, each unit was returned to power operations through the implementation of a PUP test program. This encompassed the scope of events commencing with the verification of the newly configured reactor core and terminating with the completion and review of testing at the new 100% power level. This test program was conducted with similar considerations for tests and administrative controls that were used during the Start-up Test Program. A summary discussion of the PUP Test Program, including organization and staffing, test procedures, conduct of the test program and test descriptions can be found in Chapter 14.3 of the FSAR. The critical parameters changed through power uprate were a higher reactor operating pressure and a new power-flow map, which included an increased maximum core flow. Many of the major tests in the PUP Test Program were conducted to verify acceptable performance as a result of these changes. Examples of these tests include:

- Proper operation of the HPCI and RCIC systems;

- Pressure regulator operation, including transient maneuvers;

- Feedwater system control to provide acceptable reactor water level control; and

- Recirculation system control to achieve a higher core flow.

In summary, the "uprate" of both SSES units to a higher licensed thermal power level, provided a unique opportunity for feedback on the effectiveness of design and configuration control processes. The relative lack of safety significance associated with the few deficiencies identified during this program provides added confidence that design bases have been effectively translated into plant procedures, configuration and performance. Part of the Power Uprate Program (PUP) included studies to review each affected plant system (Nuclear Steam Supply Systems and BOP Systems) to determine the system's ability to support the uprated conditions. This included review of the original

design requirements and bases for the system, the current actual plant operating conditions, and the conditions which would be expected to exist at the uprated power condition. At the end of the PUP implementation refueling and inspection outage, each unit was returned to power operations through the implementation of an extensive test program, which encompassed activities from verification of the newly configured reactor core to performance and evaluation of testing at the new power level. This test program was conducted with similar considerations for tests and administrative controls that were used during the original Start-up Test Program.

b) Improved Technical Specifications (ITS)

Technical Specifications, as Appendix A to the Operating License, set forth the limits, operating conditions, and other requirements imposed upon facility operation for the protection of the health and safety of the public. They are derived from the analyses and evaluations included in the SAR, and amendments thereto. Given this scope, development of Improved Technical Specifications (ITS) for each SSES unit required an extensive reevaluation of existing design and licensing documentation, as well as implementing procedures.

The SSES ITS project was initiated in April of 1995, and began with the development of initial review packages for each section of the SSES ITS. During the development of the initial review packages, the SSES Current Technical Specifications (CTS) were not used as the primary source of design basis information. Instead, documents including the SSES FSAR, Design Bases Documents (DBDs), calculations, controlled drawings, the NRC Safety Evaluation Report for SSES and other licensing documentation, industry standards and guidelines, etc., were used to develop the ITS. Design limits specified in the SSES ITS were certified, as were SSES ITS Bases statements. If a design limit or statement could not be certified via consistency with an approved design document, an SSES ITS open item was created and resolved by the appropriate PP&L organization.

Following the development of each SSES ITS package, the package was distributed for cross-functional review throughout the Department (e.g., Operations, Chemistry, Health Physics, Nuclear Engineering, Licensing, etc.). To ensure a complete and comprehensive review, guidelines were developed to provide different organizations with guidance on the review of the SSES ITS. Three guidelines were developed and distributed with the packages. These guidelines were as follows:

(1) Engineering Review

Engineering organizations within PP&L reviewed the SSES using a guideline to ensure that the design information contained in the SSES ITS was correct and was consistent with other design bases documents. If inaccuracies were identified, they were documented and resolved.

(2) Plant Staff Review

SSES Operations, Systems Engineering and other SSES support organizations, such as Health Physics and Chemistry, performed a review of each SSES ITS package to ensure all requirements as described in the SSES ITS and ITS Bases could be implemented at SSES. Where comments were identified, they were documented and resolved.

(3) Licensing Review

PP&L Licensing performed an independent review to verify that the SSES ITS conversion captured all technical changes and properly categorized these changes. Where problems were identified, they were documented and resolved.

Following the review of the initial SSES ITS packages, comment resolutions were incorporated into the final SSES ITS submittal. Changes were only made as a direct result of an open item resolution, a comment resolution, or a

further review and clarification of an SSES design document. It should be noted that if any discrepancies were found, the appropriate PP&L organization was contacted, a review was performed, and as necessary, change documentation issued. For those discrepancies associated with the plant design, individual Condition Reports were generated for each discrepancy to ensure that an adequate evaluation was performed and the condition appropriately corrected. There were only a few issues that resulted in the generation of Condition Reports (< 10). None of these items indicated breakdowns in design and configuration controls, or represented challenges to operability or reportability.

Following the initial review, a PORC Subcommittee was provided a new SSES ITS Review Package with comments incorporated. The PORC Subcommittee review consisted of a cross-disciplinary review of the SSES ITS and ensured that the proper PP&L reviews had been performed.

Following the PORC Subcommittee review, the full PORC reviewed the SSES ITS package and then a review was performed by the PP&L SRC (first a subcommittee review followed by presentation to the full SRC).

The development of the full SSES ITS submittal package took approximately 15 months. During this time, design changes were being implemented at SSES. To ensure the SSES ITS properly reflected these changes, two specific reviews were performed. First, all pending and recently incorporated SSES CTS changes were identified and properly reflected in the SSES ITS submittal. Second, a review by PP&L organizations was performed of all plant modifications being developed and implemented at SSES to ensure these modifications would not result in a change to the SSES ITS.

A final integrated review was performed by key members of the SSES ITS Review Team. This review was a final evaluation to assure that comments were adequately incorporated. This review also ensured that no inconsistencies existed between the SSES ITS sections. The finished package was then presented to PORC for its final approval and to SRC for its final review. A submittal to the NRC was made on August 1, 1996.

In summary, our recent submittal of Improved Technical Specifications (ITS) for NRC review and approval provided another opportunity to assess the adequacy of the design bases. The SSES ITS Project, initiated in April of 1995, essentially reevaluated the bases for system operation. The ITS Project began with the development of initial review packages for each section of the SSES ITS. During the development of the initial review package, the SSES current Technical Specifications were not used as a design source. Instead, controlled sources of design and licensing basis information were used to develop the SSES ITS. Design limits specified in the SSES ITS were verified, as were statements in the SSES ITS Bases. If a design limit or statement could not be verified via consistency with an approved design document, an SSES ITS open item was created and resolved by the appropriate PP&L organization. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that design bases have been effectively translated.

c) *Design Basis Documentation Project*

In 1992, PP&L began a design basis initiative, referred to as the Design Basis Documentation (DBD) Project, that is primarily directed at better organizing its design basis information. Because the initial design basis information for the SSES units was technically sound and well controlled, PP&L concluded that complete design basis reconstitution was unnecessary for SSES. However, when our design basis initiative (or other assessment, change or corrective action process) has indicated the need for enhancement in specific areas, PP&L has taken, and will continue to take, the necessary corrective actions. The DBD Project has generally followed the "mixed approach" presented in NUMARC 90-12. The mixed approach uses a combination of text and extensive references to document design bases. The DBD Project contains five major elements:

(1) <u>DBD Development</u>: Three DBD types were selected for the scoping process: system (e.g., HPCI), structure

(e.g. containment) and topic (e.g. equipment qualification). All potential SSES DBDs were then scoped into one of these three types. This resulted in the selection of 49 DBDs for development. The criteria which was used to select the 49 DBDs emphasized safety-related systems, NSSS systems, risk-significant systems as defined by the maintenance rule, important balance-of-plant systems, customer needs, etc.

A standard format for the DBDs was also established to include: a system overview (e.g., purpose, boundaries, subsystems, etc.); design basis requirements as defined by NUMARC 90-12 (including design basis features which implement the requirement, identification of licensing basis requirements, design basis evolution, etc.); and related references (e.g., calculations, specifications, modifications, NRC correspondence, etc.).

To date, a total of 18 of the 49 DBDs have been completed. These 18 DBDs include:

- Residual Heat Removal Service Water/Emergency Service Water
- Control Rod Drive Hydraulic System/Reactor Manual Control System
- Class 1E DC Electrical
- High Pressure Coolant Injection
- Single Failure & Separation Criteria
- Class 1E AC Electrical
- Leak Detection
- Environmental Qualification
- Diesel Generator & Auxiliaries
- Residual Heat Removal
- Reactor Protection System
- Core Spray
- Plant Design Criteria
- Reactor Core Isolation Cooling
- Standby Liquid Control
- Seismic & Hydrodynamic Loads
- Cable & Raceway
- Appendix R

(2) Development of a Screen Managed Automated Retrieval Technology System (SMARTS) Information System: A software system has been developed to provide DBD users with text-searchable electronic copies of completed DBDs as well as selected licensing basis documents. Hyperlinking capability to electronic images of most DBD references is also provided. The SMARTS system allows users throughout the department to access this information via their desktop computers.

(3) Validation of the DBDs: Each DBD is independently reviewed with the purpose of providing reasonable assurance that the design basis requirements have been identified and incorporated into the plant design, and that the requirements are consistently reflected in the physical plant and those controlled documents used to support plant design, maintenance and operations. Relative to operating, maintenance and testing procedures, the Surveillance Testing procedures are reviewed during DBD validation more often than operating or maintenance procedures. However, to the extent that operating, maintenance or testing procedures should reflect design basis requirements, the DBD Project tracks the requirement to the relevant procedure and verifies conformance with the requirement.

(4) Identification, Prioritization and Tracking of Open Items: During development and validation of a DBD, open items are identified based upon missing, conflicting or incorrect information, or any other concern which requires evaluation and response. These items are identified and tracked in accordance with written procedure. Upon identification, the open items are screened for safety significance and Condition Reports generated, if required. As a result investigations during Phase I of the CLB Project, the process was recently enhanced to require immediate generation of a Licensing Document Change Notice (see Section II) if a licensing document is affected by an open item. Each DBD contains a list of open items and associated dispositions.

(5) <u>Calculation Indexing and Scanning</u>: Approximately 23,000 calculations, which effectively represent all the design calculations for SSES, have been scanned into an optical disk storage system and re-indexed for easy retrieval via the SMARTS information system described above. (Note: 4000 of the very large piping and civil calculations only have summary sheets scanned in to refer the user to hard copy calculations.)

The results of DBD validation activities provide some indication of the effectiveness of our translation of the design basis into the plant configuration. During the validation of the first 18 DBDs, less than one dozen deficiency reports surfaced out of all the open items uncovered during the validation process. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that plant procedures, configuration and performance are consistent with the design bases.

3. Current Licensing Basis (CLB) Project

The general purpose of the CLB Project is to assure the overall health of PP&L's current licensing basis. Specific objectives include: (1) characterizing the "health" of the FSAR and taking immediate action, where warranted; (2) focusing the problem evaluation such that enhancement actions can be effectively and efficiently executed; and (3) identifying and resolving any potential programmatic/process weaknesses. The overall evaluation plan is a three-phased approach, in which the scope of one "phase" of the project is determined by the investigation results of the preceding phase. A detailed verification effort for Phase III, "Verification," was preserved as a contingency, in the event that the results of the Phase II assessment reveal that complete FSAR verification is warranted.

As part of the overall Nuclear Department Assessment Plan developed in 1995, PP&L initiated an assessment effort in February 1996 to assess the SSES FSAR relative to emerging industry issues. The completed assessment became "Phase I" of the CLB Project. The objectives of the assessment included: (1) defining the current FSAR industry issue, and (2) providing recommendations for enhancing the accuracy of the SSES FSAR.

The assessment recommended several short-term actions, possible process enhancement actions, and a scope for further assessment activities. Activities targeted towards accomplishing objective #1 included: researching general characterizations of the FSAR accuracy concern by the regulator; monitoring NRC/industry reports concerning FSAR inaccuracies during the assessment; and monitoring the results of NRC inspections of SSES, conducted in accordance with NRC guidance regarding FSAR assessment objectives. Objective #2 was satisfied by the following activities: development of a categorization process for application to industry events and apparent FSAR discrepancies; quick-turnaround data-gathering and analysis of outstanding FSAR discrepancies identified within the Department; review of regulatory requirements applicable to FSAR maintenance (e.g., 10 CFR 50.59, 10 CFR 50.71(e)); and evaluation of the processes used by PP&L to ensure adherence to these requirements.

The Phase I assessment revealed that a number of apparent FSAR discrepancies had been identified via DBD Open Items (see discussion above), and remained outstanding. Additionally, a number of Licensing Document Change Notices (LDCNs) required expedited processing. The evaluation of the processes established to maintain the SSES FSAR revealed a number of opportunities to formalize or strengthen internal process controls used to ensure compliance with regulations pertaining to FSAR content. Specific recommendations for process improvements addressed the following areas: administrative details regarding the processing of LDCNs and FSAR revisions; 10 CFR 50.54 reviews of the Emergency Plan, Security Plan and QA Program; clarification of 10 CFR 50.59 applicability; linkages to plant change processes and maintenance of the FSAR; and commitment management.

Phase II of the project commenced in June 1996, and as of this writing, is ongoing. The two primary areas of emphasis include: (1) executing near-term actions to address the recommendations of Phase I; and (2) performance of a broad "scoping" assessment of the FSAR, in order to determine the level-of-effort necessary to perform detailed verification of the FSAR, should this prove necessary. Results of the Phase II assessment, and recommendations regarding the need for additional FSAR verification will be provided as inputs in developing the scope of Phase III.

One of the major emphases of the Phase II project was to fully resolve the "apparent" FSAR discrepancies identified in the Phase

I report. The majority of these discrepancies were identified via DBD preparation and validation activities. As a conservative measure, a Condition Report was generated to identify the immediate actions recommended by the Phase I assessment. Resolution of the apparent discrepancies entailed investigation of each item by senior engineers, and determination of the appropriate action. As of December 1996, 140 "apparent" FSAR discrepancies have been dispositioned, with the following results:

- 74 required no action (i.e., were not in fact discrepancies between the FSAR, design and plant configuration);
- 53 were resolved directly via a Licensing Document Change Notice and did not necessitate generation of a Condition Report;
- 9 required further investigation by other elements in the organization, but did not necessitate generation of a Condition Report; and
- 4 items resulted in Condition Reports (note that the CRs generated did not challenge operability or reportability).

Additionally, process improvements are in development (see Section II), and LDCN incorporation was expedited to ensure all updates were incorporated. Revision 49 to the FSAR was submitted in May 1996, Revision 50 in August, 1996, and Revision 51 is planned for the first quarter 1997.

It should be noted that this is an in-process list. Final Phase II activities include an "aggregate analysis" of the apparent discrepancies listed above, the results of FSAR scoping reviews, any related CRs generated independent of the project, and industry events within the assessment period. The purpose of this analysis is to determine whether any generic implications or process issues exist. The report of Phase II activities will recommend short-term actions, process enhancements, and recommendations for the scope of Phase III activities. Preliminary observations from Phase II activities include: (1) no safety significant discrepancies have been identified in the population of apparent FSAR discrepancies; (2) FSAR scoping reviews have revealed that certain FSAR sections contain outdated information; and (3) Department personnel have a heightened awareness of FSAR accuracy requirements due to supplemental refresher training and reinforcement from supervision and management.

Note that during the performance of Phase II, the Nuclear Energy Institute (NEI) issued Industry Initiative No. 96-05. The initiative provides guidance "for performing self-assessment of the adequacy of programmatic controls for maintaining the licensing basis in order to identify missing or incorrectly applied programmatic elements that could lead to licensing basis differences." In support of the industry initiative, PP&L is conducting additional focused reviews, in order to provide NEI with data for compilation.

Although actions are currently ongoing, the scope of Phase III of the CLB Project is not anticipated to be finalized until mid-1997 to ensure that all scope inputs are considered. The scope for Phase III will address the design basis concerns identified in the NRC October 9, 1996 request for design basis information. Specific "feeders" to the scope of Phase III include: (1) the results of Phase II of the CLB Project; (2) recommendations from the team charged with development of the response to your October 9, 1996 request; and (3) results of implementing NEI Initiative 96-05. These feeders will include process improvements, short-term actions and recommendations for additional assessment.

In summary, the ongoing CLB Project, initiated in early 1996, will further evaluate the design bases issues raised in the NRC request for information. PP&L has already identified discrepancies and improvement opportunities as a result of ongoing scoping reviews of the FSAR, and expects that more will be found. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that design bases have been effectively translated into plant procedures, configuration and performance. Nevertheless, PP&L will perform an aggregate analysis of all identified discrepancies as a means of identifying any generic process implications. PP&L will continue to evaluate these items as they are identified, and take appropriate action to resolve them.

E. Conclusion

Recognition of the rigor of the start-up process is an important consideration, and a key to our confidence that fidelity has been maintained between the design bases and the plant configuration. Specifically, the turnover and start-up testing activities provided reasonable assurance that the original design was consistent with technical standards, that the design basis information provided a sufficient foundation from which to control changes, and that the plant configuration reflected the design basis information. Consistency between design bases and the plant configuration was also confirmed through an independent design verification of the feedwater system in support of initial plant licensing.

Furthermore, the involvement of PP&L personnel throughout the start-up process established an understanding of the importance of consistent translation of design bases into plant procedures and the plant configuration. As described in Section II, prior to the start-up, PP&L recognized the need to make the transition between the design processes utilized during the design and construction phases to a set of controls more appropriate for changes made during the operational phase. This entailed management of design and configuration changes using the very important concepts of "as-engineered" and "as-built" documentation. Refer to Section II, III and IV for details.

From this baseline, PP&L has maintained ownership of the design, operation and maintenance of the plant using qualified and experienced personnel, and appropriate procedures. The knowledge-base of our personnel has continued to grow, in turn allowing PP&L to retain ownership of design, rather than relying heavily upon contractors for design work. Additionally, the QA framework was expanded upon by a set of Supplemental Procedures to define and facilitate the transition between construction and operational phases of SSES, and ultimately by the development of the PP&L Operational QA Program.

Within the framework of the quality assurance program, our processes have continued to evolve. As described in Section II, prior to start-up, PP&L recognized the need to make the transition between the design processes utilized during the design and construction phases to a set of controls more appropriate for changes made during the operational phase. Section II of the attachment describes PP&L's current integrated set of design and configuration control processes. These processes include

mechanisms for considering the design bases during design, transforming a design change into information describing the configuration of the plant, and then finally for implementing design changes into the actual physical configuration of the plant. Section III addresses the questions of how design bases were originally translated into plant procedures, and how design changes are currently translated into plant procedures. Section IV addresses the processes used to translate the design changes into the plant configuration. Section V addresses our mechanisms for problem identification and corrective action. These are the processes we use to identify anomalies in the design and configuration controls, and assure that processes evolve through lessons learned from operating experiences.

Audits and assessments (internal and external) have provided evidence supporting our confidence in the effectiveness of our processes and programs. Note that in some cases, these reviews have identified various weaknesses related to control of the design bases or consistency between the design bases and the physical plant. Upon identification, these items are entered into the corrective action process where they are evaluated for safety significance, operability, and reportability, and where appropriate, corrective actions are implemented.

PP&L has also undertaken major projects which provided unique opportunities for feedback on the effectiveness of design and configuration control processes used to translate design bases. The relative lack of safety significance associated with the few deficiencies identified during these broad reviews provides added confidence that design bases have been effectively translated into plant procedures, configuration and performance.

Thus, PP&L has reasonable assurance that design bases requirements are consistent with the plant procedures, configuration and performance. This is based in part on our confidence in the processes themselves, as well as on feedback which provides evidence of effective process implementation in translating the original design, and changes to the design over the operating history of SSES. Our reasonable assurance is also based upon our confidence in our quality management systems. As described above and within the attachment, our QA program has provided the framework upon which we have developed and refined our design "translation" processes, throughout the design, construction and operational phases of SSES. This program and supporting processes have evolved over the years, due in part to our QA oversight and assessment functions. We have also accrued benefits from the expanding knowledge-base of our personnel over time, in implementing these processes.

In summary, our reasonable assurance that plant procedures, configuration and performance are consistent with the design bases is based upon the following key points:

- Establishment of a solid original design baseline, developed within a quality assurance framework, using qualified personnel.

- Implementation of integrated change management and corrective action processes within a quality assurance framework, using qualified personnel.

- Feedback on the effectiveness of the processes in translating design bases through continuing activities such as oversight and assessment (internal and external), as well as through unique opportunities provided via major design review projects.