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Company

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

"REQUEST FOR INFORMATION PURSUANT TO 10CFR50.54(f) REGARDING ADEQUACY AND AVAILABILITY OF DESIGN BASES INFORMATION"
HOPE CREEK GENERATING STATION
DOCKET NO. 50-354

Gentlemen:

By letter dated October 9, 1996, the Nuclear Regulatory Commission requested PSE&G to submit information that will provide the NRC with added confidence and assurance that our nuclear facilities are operated and maintained within the design bases and any deviations are reconciled in a timely manner. As requested, PSE&G is submitting our response to the 10CFR50.54(f) letter within 120 days of its receipt. The two attachments provide PSE&G's response to the subject letter for the Hope Creek Generating Station.

PSE&G agrees it is essential to have programs in place that ensure that Hope Creek is configured and operated in accordance with its design bases. Further, PSE&G is committed to assuring that there are programs and procedures in place that adequately support the maintenance of the design bases. PSE&G has undertaken a number of specific validation efforts to provide reasonable assurance that the design bases are maintained and plans to conduct additional reviews to provide further assurance. In addition, the new Nuclear Business Unit (NBU) culture, which focuses on the identification of problems through the use of critical self assessments and a questioning attitude, will ensure problems are resolved in an effective and timely manner in accordance with our revamped Corrective Action Program.

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The detailed responses to each of the five specific information requests are provided in Attachment 1. Attachment 1 also includes a detailed Table of Contents and Executive Summary for your convenience. It should be noted that while the discussion of processes and programs contained herein provide an accurate description of their present state, PSE&G will continue to revise them in accordance with approved

revision processes as further enhancements are identified, without modifying this response. In order to eliminate any ambiguity with regard to commitments contained in this response, Attachment 2 describes the PSE&G specific commitments related to this request.

Based on the detailed information provided herein, PSE&G has concluded that there is reasonable assurance that the Hope Creek facility will be operated in accordance with its design bases. The future actions discussed in Attachment 2 will provide additional assurance of compliance with design bases. If there are any questions regarding this information, we will be pleased to discuss them with you.

Sincerely,

E C Simpson

Attachments(2)

C Mr. Hubert J. Miller, Administrator - Region I
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Mr. D. Jaffe, Licensing Project Manager
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REF: LR-N970093

STATE OF NEW JERSEY)

) SS.

COUNTY OF SALEM)

E. C. Simpson, being duly sworn according to law deposes and says:

I am Senior Vice President - Nuclear Engineering of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning the Hope Creek Generating Station, are true to the best of my knowledge, information and belief.

EC Simpson

Subscribed and Sworn to before me

this 11th day of February, 1997

Donna J Willadsen

Notary Public of New Jersey

My Commission expires on June 2000

HOPE CREEK GENERATING STATION

Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

ATTACHMENT 1

HOPE CREEK GENERATING STATION
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Executive Summary

This attachment provides PSE&G's response to the NRC letter of October 9, 1996, in which information was requested pursuant to 10CFR50.54(f) regarding the adequacy and availability of design bases information at the Hope Creek Generating Station. Specifically, the NRC's letter requested the following information:

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

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

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

D. Processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, action to prevent recurrence, and reporting to the 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 response was prepared by a team of several dedicated, experienced personnel. The team focused on review of (1) processes that control the design and configuration of the plant, (2) numerous validation activities and various independent assessments, both internal and external, to assess the translation of design bases into plant configuration and operations, and (3) the plant corrective action process. In addition to the requested information, a separate discussion is provided that gives reasonable assurance that the current design bases at Hope Creek is both adequate and accurate.

The "Response to Question A" section describes the major processes currently used to maintain plant design and operations consistent with the design bases. These processes include the configuration control process, design and configuration change process, document control process, 10CFR50.59 safety evaluation process, Technical Specification amendment process, and the 10CFR50.71(e) UFSAR update process. "Response to Question A" also

describes human performance initiatives to improve personnel performance relative to understanding and sensitivity to design bases issues.

The "Response to Question B" section describes the various validation reviews and independent assessments performed over the past several years which provide us reasonable assurance that design bases information has been adequately translated into plant procedures. These activities include reviews of a sample of Hope Creek procedures, the Technical Specification surveillance improvement project, and the configuration baseline document validation program. The "Response to Question B" section also summarizes key independent assessment findings, both strengths and weaknesses, to evaluate the effectiveness of process controls discussed in "Response to Question A".

The "Response to Question B" section concludes, based upon results of validation reviews and independent assessments, that there is reasonable assurance that the Hope Creek design bases have been translated into operating, maintenance, and testing procedures such that continued plant operation can be supported. PSE&G has confidence that the corrective action program, coupled with additional planned procedure reviews (discussed in Attachment 2), will identify any existing discrepancies between procedures and the design bases. When discrepancies are identified, appropriate actions will be taken in accordance with the corrective action program and the plant Technical Specifications.

The "Response to Question C" section describes validation efforts and independent assessments related to design bases consistency with plant configuration. These activities include the Hope Creek Independent Design Verification Program (IDVP), Readiness Assessment Team Inspection (RATI) 96-80, the Electrical Distribution System Functional Inspection, and the Service Water Operational Performance Inspection. The "Response to Question C" section also summarizes key independent assessment findings, both strengths and weaknesses, to evaluate the effectiveness of process controls discussed in "Response to Question A".

In "Response to Question C", PSE&G concludes it has reasonable assurance that structure, system, and component configuration and performance are consistent with the plant design bases. As additional design bases or configuration issues are discovered through continuing review activities, they will be addressed and resolved in accordance with our corrective action program and plant Technical Specifications.

The Hope Creek corrective action program is described in "Response to Question D", including the Action Request process, operating experience feedback process, 10CFR21 process, and the employee concerns program. The corrective action program was substantially revised in the 1994/1995 time frame as a result of inspection findings which questioned the effectiveness of the previous system. Throughout 1996, numerous assessments and inspections were conducted by internal and external organizations, including the USNRC. The new corrective action program, implemented in July 1995, was demonstrated to be generally effective, including the ability to identify and correct design related issues. While improvements are still needed relative to the timeliness and effectiveness of corrective actions, the new corrective action program has been effective in resolving safety significant issues that have arisen.

In "Response to Question E", PSE&G states that sufficient information has been presented to conclude there is reasonable assurance the Hope Creek Generating Station is configured, operated and maintained within its design bases, such that continued plant operation can be supported. As additional design or design bases related issues are discovered through continuing review activities, they will be addressed and resolved in accordance with our corrective action program and plant Technical Specifications.

In general, PSE&G believes its corrective action program, document control and retrievability, and operating procedures are strong, but that UFSAR consistency can continue to be improved and engineering-staff design bases knowledge can be improved. To address known weaknesses and provide additional assurance of compliance with the design bases, a comprehensive design bases review project will be undertaken over the next two years, as further discussed in Attachment 2.

In Attachment 2, PSE&G outlines our program to continue correcting deviations or deficiencies in design bases documentation, as they are identified, and improving the consistency of the information with plant configuration and procedures. A formal submittal will be placed on the docket within 60 days of this letter, providing program details and schedules.

In general, we intend to conduct the proposed activity in system teams comprised of our system managers, system design engineers, system SROs, and other support personnel as required. The system teams will conduct further adequacy reviews of design bases information and documentation, and the translation of the design bases into operations, in order to both gain knowledge and the ability to revise and update the UFSAR and CBDs. As in the case of our

1996 FSAR Project, systems will be classified as safety analysis, risk significant, risk important, or other. The level and extent of reviews will vary depending upon system classification.

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Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

INTRODUCTION

The Hope Creek Generating Station (HCGS) is a General Electric Boiling Water Reactor (BWR) with a Mark 1 style containment. Hope Creek is located on southern part of Artificial Island on the east bank of the Delaware River in Lower Alloways Creek Township, Salem County, New Jersey. Bechtel Construction Inc. was the Architect Engineer for this project. Hope Creek was granted a construction permit November 4, 1974, and it's low power license on April 11, 1986. Initial Criticality was achieved on June 28, 1986, and after completion of low power testing the full power license was granted July 25, 1986. The HCGS declared commercial operations on December 20, 1986.

HCGS submitted the Final Safety Analysis Report (FSAR) to the NRC to support the full power license submittal. The FSAR conforms to the format and content requirements of Regulatory Guide 1.70, Revision 3, in that the content is in accordance with the NRC Standard Review Plan (SRP)-NUREG-800. The SRP is part of a continuing regulatory standards development activity that documents current methods of review, and more completely identifies the specific NRC requirements that are germane to each review topic (including TMI related action items), and to fully describe the NRC acceptance criteria for each review topic.

By letter dated October 9, 1996, the NRC issued to all licensees a request for information pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information. The NRC specifically requested licensees to provide the following information:

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

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

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

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

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

The NRC further requested that, in responding to items (a) through (e), licensees indicate whether any design review or reconstitution programs have been undertaken, and if not, provide a rationale for not implementing such a program.

This attachment provides PSE&G's response. The response was prepared by a team of several dedicated, experienced personnel who interfaced with several organizations in the Nuclear Business Unit. The team focused on (1) design, configuration control and corrective action processes, (2) effective translation of design bases information into the plant configuration and operations, and (3) independent assessments of configuration or operations in accordance with the design bases.

The organization of this response follows the order shown above, and is further detailed in the Table of Contents. In addition to the requested information, a section entitled "Adequacy of Design Bases" is added to provide reasonable assurance that Hope Creek's initial design bases were adequate and accurate. This provides the cornerstone for the rest of the response. In addition, since only 10 years have elapsed from the time of initial licensing of Hope Creek, the initial licensing submittal will be the baseline used for this request for information. For the validation efforts presented in the response to questions (b) and (c), a review back to the initial licensing was conducted. However, due to the recovery/restart efforts currently on-going at Salem, a number of process changes and enhancement, that effect both the Salem and Hope Creek facilities, have been made to the configuration control process. Therefore the assessments used to validate or assess the process effectiveness only goes back two to three years.

The "Response to Question A" section provides a description of the major processes currently used to maintain plant design and operations consistent with the design bases. These descriptions are summaries of our processes intended to convey a sense of how they work, and are not intended to provide all procedural details; and the process is not static and may change in the future. Included in these descriptions are process and organizational interfaces, and the elements intended to satisfy 10CFR50.59, 10CFR50.71(e) and support 10CFR50, Appendix B, requirements.

The "Response to Question B" section provides the rationale to our reasonable assurance that the design basis is adequately translated into the operation of the

facility. This section will discuss the fidelity of the implementing process used to control and maintain the design bases, such that it will address the procedure control process and other process such as work control. This section will present major validation efforts used to provide reasonable assurance that the design bases have been adequately translated and maintained into the implementing controls. In addition, this section will build upon the process discussion presented in the section entitled "Response to Question A" by highlighting the effectiveness of the processes used to translate the design bases into operation of the facility.

The "Response to Question C" section describes validation efforts and independent assessments associated with design bases consistency with plant configuration and operation. The purpose of this section is to provide the rationale for our reasonable assurance that system, structure, and component (SSC) configuration and performance are consistent with the design bases. As with the response to question (b), this section will expand on the processes described in the section entitled "Response to Question A" by identifying and demonstrating how key attributes of configuration control are fulfilled at HCGS.

The Hope Creek corrective action program is described in the section entitled "Response to Question D". This section provides a brief overview of the operation of the Action Request Process (issue identification and prioritization process) and the Corrective Action Program including conditions adverse to quality, operability determinations, reportability determinations and action closure and the 10CFR21 process. A summary of how the Operating Experience Program adds external information to the process is included. The important role of the Employee Concerns Program for identifying internal issues for entry into the Action Request Process is also described. In addition, information is provided on the effectiveness of PSE&G's implementation of these programs.

The "Response to Question E" section is provided as an overall conclusion to the response. The effectiveness of the process is more appropriately addressed in the individual responses to questions' b, c, and d.

The requirement to address the performance of a design review or reconstitution is addressed in the "Response to Question C" section, by describing the Configuration Baseline Document project that was conducted in the 1990 timeframe. Future activities are outlined in Attachment 2.

HOPE CREEK GENERATING STATION

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ADEQUACY OF DESIGN BASES INFORMATION

The purpose of this section is to provide a brief overview of our rationale for reasonable assurance that design basis information at Hope Creek is adequate. The NRC's request for information pursuant to 10CFR50.54(f) focuses on translation of design bases information into the actual plant configurations and plant procedures. As further described in this section of the response, PSE&G's rationale for translation adequacy depends upon results of various validation reviews and independent assessments of the configuration control process. Many of these reviews and assessments not only provide reasonable assurance of adequate translation of information into the configuration and procedures but also provide reasonable assurance of design basis information adequacy, as further discussed below.

Design bases information is defined in 10CFR50.2 as that information which identifies the specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values are typically derived in analyses or calculations or may be restraints derived from generally accepted practices for achieving functional goals, such as limits imposed by specifications, industry standards, manufacturer's technical manuals or data sheets, etc..

Hope Creek Generating Station was one of plants licensed in the early- to mid-1980s which received an independent design verification program (IDVP). One nuclear architect-engineer firm (Sargent & Lundy) conducted a review of the design documentation of the design architect-engineer firm (Bechtel). Like the safety-system-functional inspections (SSFIs) that followed, the IDVP involved a deep-vertical-slice review of three safety-related sample systems.

The S&L team contained a large number of multi-disciplinary engineers (mechanical, electrical, instrumentation and controls, and civil/structural). The team evaluated not only the design of the specific sample systems (HPCI, ADS, and SACS) but also the design of buildings containing the systems and common hazards and design considerations, such as internal flooding, missile protection, pipe whip, jet impingement, tornado missiles, etc. The team evaluated calculations, specifications, drawings, vendor information in order to draw conclusions relative to design accuracy, reasonableness and adequacy of design techniques, and the performance of the design process. The IDVP also involved extensive system and plant walkdowns to verify as-built configuration conformance to the design and design assumptions.

The S&L team concluded that the IDVP and on-going design verification activities provided reasonable assurance that the overall design of Hope Creek was technically adequate and conformed to FSAR commitments and regulatory requirements. Conclusions were not limited to the three systems (and appropriate buildings) because the design sample was large enough and the review was in sufficient depth and detail to draw overall conclusions relative to effectiveness of the design process. Hence, results and conclusions from the sample were extrapolated to the overall Hope Creek configuration because the same effective design process which successfully implemented design commitments in the sample systems and buildings, was used throughout the entire power plant.

The IDVP process included regular oversight visits by a team of NRC engineers and consultants. Based on their own reviews, the NRC oversight team concluded the Sargent & Lundy assessments were in sufficient breadth and depth to draw valid conclusions as to the process effectiveness. Sargent & Lundy concluded, and the NRC team concurred, that design requirements had been adequately implemented. Also, the teams concluded that, while additional reviews would likely uncover similar deficiencies, the nature and extent of the deficiencies were not such as to preclude system safety functions.

Considering the fact that Hope Creek design was conducted at a time where there was more focus on documentation and considering the activities and results of the IDVP, the Hope Creek design bases, at the time of commercial operation, was better documented and better defined than typical design bases documentation at older, non-IDVP units. In spite of the fact that the Hope Creek design baseline was considered above average in the mid-1980s, PSE&G undertook an extensive effort to compile design basis information in 1991 when it began preparation of Configuration Baseline Documents (CBDs). The program continued until 1994, with a total of 29 CBDs completed. The 29 completed CBDs included primary safety systems such as RHR, HPCI, RCIC, service water, and emergency diesel generators. The CBDs identify the system or structure design bases. Equally as important as the specific design values themselves, the CBDs also provide reference to the various source documents (calculations, analyses, specifications vendor drawings, etc.) from which the design bases values are derived. At the time they were prepared, the CBDs were peer reviewed and approved, such that they met ANSI Standard design input document requirements.

Design bases information was collected, compared to the as-built configuration, and recorded. Where discrepancies were noted, discrepancy forms (DEFs) were

prepared, evaluated, and resolved or scheduled for subsequent resolution. As such the CBD project provided an additional base-line for the Hope Creek design, beyond that of the IDVP.

Where design calculations were not available to document a particular system design feature or parameter, discrepancy evaluation forms (DEFs) were generally prepared which subsequently resulted in calculations or other engineering activities to document or otherwise confirm the design requirement. Where existing calculations were available, the CBD project identified these calculations and determined that they had been verified and approved in accordance with existing procedures. The project then used calculation output, where appropriate, in preparing the CBD. The project did not review these calculations for technical adequacy nor did it attempt to verify all input and assumptions as being appropriate.

Consequently, while the CBD project provides additional assurances that system design bases meet requirements and commitments, the type of comprehensive design reviews typically associated with safety system functional inspections or similar design reviews or inspections were not conducted. However, when combined with the results of independent inspections and other assessments which provide reasonable assurance of design bases technical adequacy, the CBD project fills an important role in that for the 29 systems completed (1) necessary design information was located to its source (2) missing information was identified and obtained or reconstituted and (3) consistency with design commitments was evaluated.

In view of the above, PSE&G has reasonable assurance that the Hope Creek design bases is adequate to assure systems, structures, and components can perform their intended safety functions. This reasonable assurance is based upon:

- The Hope Creek IDVP provided a design baseline in accordance with the higher documentation standards in effect for the newest plants in the industry.
- The CBD project further collected and verified design bases information. Deficiencies are being addressed or were resolved through the DEF closure process.
- The configuration control process, including the corrective action process, has been demonstrated effective by numerous reviews and assessments. The FSAR Project and CCW system SSFI, both conducted at Salem Station in 1996, provide reasonable assurance of the effectiveness of the Hope

Creek configuration control processes, since they are the same processes in effect at Salem.

The remainder of this section of PSE&G's response to the NRC's information request will address translation of design bases information into plant configuration or plant procedures.

RESPONSE TO QUESTION A

Description of the engineering design and configuration control processes, including those that implement 10CFR50.59, "Changes, Tests and Experiments," 10CFR50.71(e) "Maintenance of Records, Making of Reports", and Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants ....".

I. Overview

Public Service Electric and Gas (PSE&G) has many design and configuration control processes, including processes that implement 10CFR50.59, 10CFR50.71(e) and Appendix B to 10CFR50. These processes are integrated into the daily operations and maintenance of the Hope Creek Generating Station. Changes to plant design and configuration are controlled by use of approved procedures, and involve trained and qualified personnel at various levels of the organization. The majority of these procedures are common to both the Hope Creek and Salem Stations.

The processes described below are currently used to maintain plant design and operations consistent with the design bases. These descriptions are summaries of our processes intended to convey a sense of how they work, and are not intended to provide procedural details; and the processes are not static and may change in the future. Included in these descriptions are process and organizational interfaces, and the elements intended to satisfy 10CFR50.59, 10CFR50.71(e) and support 10CFR50, Appendix B, requirements.

II. Appendix B to 10CFR50

Appendix B to 10CFR50 establishes quality assurance requirements for the design, construction and operation of those structures, systems, and components (SSCs) that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

The Quality Assurance Program (QAP) is an essential part of the PSE&G commitment to safe and reliable operation of the Hope Creek Power Plant. The QAP is described in the Hope Creek Final Safety Analysis Report and meets the requirements of 10CFR50 Appendix B. The QAP provides control over

independent and combined activities affecting the quality of identified SSCs consistent with their importance to safety.

10CFR50, Appendix B, is divided into 18 criteria relative to Quality Assurance which are to be applied to the design, fabrication, construction and testing of the SSCs. PSE&G has implemented processes to meet each of these criteria. These processes are driven by PSE&G Nuclear Business Unit (NBU) procedures. The design and configuration control processes described below incorporate the requirements of Appendix B to 10CFR50 applicable to those activities.

Specific criteria from 10CFR50, Appendix B, which apply to the design and configuration control processes include: Design Control; Procurement Document Control; Instructions, Procedures and Drawings; Document Control; Test Control; and Quality Assurance Records. These processes are integrated by procedure into the activities related to support the facilities. Other 10CFR50, Appendix B, requirements related to program implementation, including those related to materials and equipment control and usage, and the Corrective Action Program are discussed in the "Response to Question D" below.

A comprehensive system of planned and periodic audits ensures the continuing effectiveness and adequacy of the design and configuration control processes. These audits review program compliance with 10CFR50, Appendix B, as well as effective implementation of the program elements.

III. Description of Design and Configuration Change Processes

PSE&G has established processes for design and configuration changes to the Hope Creek Station. These processes are intended to ensure that:

1. Design and configuration change activities receive the required analyses, evaluations, and reviews,

2. Design information is correctly incorporated into the final design, and

3. Configuration control is maintained during the change activity.

Proper implementation will ensure that design and configuration changes are in compliance with the plant design and licensing bases, and that documentation is appropriately reviewed to reflect design and configuration changes.

The processes that govern permanent and temporary changes to plant design and configuration at the Hope Creek Station are:

A. Control of Design and Configuration Change, Tests and Experiments (NC.NA-AP.ZZ-0008(Q)),

B. Minor Modification Process (NC.NA-AP.ZZ-0017(Q)),

C. Control of Temporary Modifications (NC.NA-AP.ZZ-0013(Q)), and

D. Nuclear Fuel Procurement and Core Control (NC.NA-AP.ZZ-0072(Q)).

A brief description of each of the above processes follows.

A. Control of Design and Configuration Change, Tests and Experiments (NC.NA-AP.ZZ-0008(Q))

Procedure NC.NA-AP.ZZ-0008 (Q), "Control of Design and Configuration Change, Tests and Experiments," (NAP-8) establishes a uniform method for controlling design and configuration changes, test, and experiments. The process is common to the Hope Creek and Salem Stations.

The NAP-8 process has the specific purpose of ensuring that proposed design changes are made in accordance with assumptions in the plant design and licensing bases, and that impacted documents (including drawings, calculations, procedures and databases) are identified and revised in an appropriate manner. Supporting procedures referenced throughout this process are designated to ensure that the appropriate requirements are understood by those developing, implementing and closing out a design or configuration change, test or experiment.

Changes controlled by the design and configuration change process fall into the three major categories of design changes, configuration changes, and tests and experiments as discussed below.

Design Change

A design change is a change to the system design that affects the functional performance or parameters of plant SSCs. Design changes, once implemented, require an update of the fundamental configuration documentation to reflect the revised as-built condition. Design changes are classified as:

1. A "standard design change" which implements a new design for such reasons as improved safety, increased capacity, or to meet new regulatory requirements.

2. An "as-built" design change, which updates existing configuration documentation, including the UFSAR, to be consistent with actual plant design following a documented design analysis.

Configuration Change

A configuration change is a change to the fundamental configuration documentation that does not affect the design bases functional performance or parameters of plant SSCs, or is typographical in nature. The four types of configuration changes are:

1. The "equivalent replacement" configuration change which involves replacement of parts or components with ones of equivalent functions, or non-design computer software changes. These component replacements may be due to obsolescence, unavailability, or lack of reliability of specific components. This change type documents the equivalency of form, fit and function.

2. The "generic equivalent replacement" configuration change pre-engineers the acceptability of whole components or piece parts as alternative equivalent replacements. Installation is performed under the work control process. This change type permits configuration control of databases as each alternative equivalent replacement is installed.

3. The "document only" configuration change is used to correct drafting or editorial errors, correct errors of omission, correct discrepancies between documents, and to revise component bill of materials (BOMs) and vendor documents to identify parts number changes due to obsolescence.

4. The "engineering change authorization" configuration change implements changes to SSCs that do not affect the functional performance or parameters described in the design bases.

Tests and Experiments

Tests and experiments are not permanent design or configuration changes to the plant, although they may involve installation of temporary equipment for their conduct. A test is a controlled set of plant operations intended to verify that

systems or components function in accordance with predetermined specifications. An experiment is a controlled set of plant operations intended to establish system or component characteristics or values not previously known.

Tests and experiments affecting configuration are controlled and documented by a change package. Design change package control procedures require incorporation of post-modification test procedures to confirm the achievement of the design change intent, and identification of any changes to facility surveillance or operating procedures necessitated by the change. If necessary, experiments, e.g., core reload physics parameter determinations, are controlled by operational procedures reviewed and approved in accordance with procedure NC.NA-AP.ZZ-0001(Q), "Nuclear Procedure System," (NAP-1).

Change Package Details

Plant configuration control is maintained during implementation of a design or configuration change through issuance of change packages, namely Design Change Packages (DCPs) and Configuration Change Packages (CCPs). The format for change packages is prescribed for each type of change in the applicable Design Engineering procedures. The change package content varies according to the type of change and its scope.

From the perspective of design controls, development of a standard DCP is an important part of the process. Typically, the process will involve consideration of the information discussed below. Other types of change packages, e.g., as-built design change, equivalent replacement configuration change, generic equivalent configuration change, document only configuration change, or engineering change authorization, will contain some or all of the described elements, as required by the applicable Design Engineering procedures as they are applicable to that specific class of change. The format of a standard DCP, which is the most comprehensive class of change, includes:

1. General

The Forms Section of the package includes documentation of approvals and revisions, an Interface Record, an Executive Summary, a Station Department Change Package Checklist, and a turnover to operations checklist.

The Interface Record documents that the discipline or specialty areas and associated interfaces have been adequately addressed and mandatory station department and installer reviews have been obtained. The Station Department

Change Package Checklist identifies affected station procedures or databases, and update requirements for these items.

Change packages are subject to the procedurally controlled peer review process. Procedure NC.DE-AP.ZZ-0009(Q), "Peer Review," (DEAP-9), establishes controlled and standard guidelines for performing a peer review of design documents in accordance with Regulatory Guide 1.64. Peer review is the process of ensuring that each design output document satisfies the specific design inputs and source criteria, and it serves to identify and correct deficiencies in the design documents. The review is performed, at a minimum, using a line-by-line check.

2. Engineering

The Engineering section includes documentation of the supporting engineering and design analysis. This section includes:

Design Bases/Input (NC.DE-AP.ZZ-0001(Q))

Procedure NC.DE-AP.ZZ-0001(Q), "Design Bases/Input," (DEAP-1) establishes a method for identifying design considerations and design input used in the preparation of design documents. This procedure is used by the change package preparer to identify the significant design inputs used in the preparation of the design document as required by ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants," under our commitment to Regulatory Guide 1.64. The procedure ensures a complete record and an efficient means of retrieval of design document information.

Specialty Review (NC.DE-AP.ZZ-0007(Q))

Procedure NC.DE-AP.ZZ-0007(Q), "Specialty Reviews," (DEAP-7) establishes guidelines for performing a review of design documentation to ensure that effects of the change on specific technical and engineering programs are considered.

The Specialty Review procedure (DEAP-7) uses checklists to determine potential impact of the change on specific technical and engineering programs. The checklists include fire protection, environmental qualification of electrical equipment/devices, mechanical equipment qualification, pipe stress analysis, seismic qualification, and the Inservice Inspection Program. In addition, security, Cold Shutdown From Outside Control Room (GDC-19), Inservice Testing and Valve Programs, Regulatory Guide 1.97 variables (Post Accident Monitoring), environmental design criteria, nuclear fuel design impact, Probabilistic Risk

Assessment (PRA), Station Blackout, setpoint calculation, and the Erosion/Corrosion Monitoring Program are also included.

The change package preparer determines the specialty reviews that are required for the design change. The programmatic specialist reviews the change package to ensure that applicable programmatic requirements are met, and that sufficient information is available to update the applicable program.

Design Analysis

As defined in procedure NC.DE-WB.ZZ-0001(Q), "Standard Design Change Workbook One," the Design Analysis section summarizes methods of analyses and discusses how the change meets the requirements and design impacts identified in the Design Bases/Input section. It includes discussions related to system performance requirements, component design conditions and constraints, and how safety requirements are satisfied. The design analysis addresses exceptions or deviations from industry/vendor standards and assumptions.

10CFR50.59 Applicability Review or Safety Evaluation

Design Change Packages require screening for 10CFR50.59 applicability as described in procedure NC.NA-AP.ZZ-0059(Q), "10CFR50.59 Reviews and Safety Evaluations," (NAP-59). If the change is within the scope of 10CFR50.59, a safety evaluation is performed prior to implementation of the design or configuration change. The safety evaluation determines if an unreviewed safety question (USQ) exists.

The 10CFR50.59 Applicability Review(s) and Safety Evaluation(s), if required, are included in this section of the change package. The Station Operations Review Committee (SORC) is required to approve the change package if a 10CFR50.59 Safety Evaluation is required. These reviews are performed in accordance with the NAP-59 process, which is discussed in section VI below.

Design Verification (NC.DE-AP.ZZ-0010(Q))

Procedure NC.DE-AP.ZZ-0010(Q), "Design Verification," (DEAP-10) establishes guidelines for performing independent design verifications of design documentation to meet the requirements of Regulatory Guide 1.64. Design verification is performed on documentation for Q-listed items and non-Q-listed items that are considered to be important to safety, resulting from Technical Specification (TS) changes, or are changes to previously verified designs. The

design verifier determines the method, extent and depth of verification; performs verification of design documentation, specialty reviews and safety reviews. The design verifier also documents the acceptability of the design documentation.

3. Modification and Testing

As defined in procedure NC.DE-WB.ZZ-0001(Q) "Standard Design Change Workbook One," the Modification and Testing section contains the information required by the installer in the field to complete the change. This section includes: a Modification Details section which provides the requirements to assure proper modification installation; a Testing section which includes operational, functional and special testing requirements; a Modification Documents section which lists the documents impacted by the change that are required to support installation; and the Equipment, Services and Materials section which lists material required to install the modification. If required, a Code Job Package is also included in this section.

4. Close-out

The Close-out section contains items required to ensure that the change has been completed in accordance with the proposed design, and that documents/databases have been updated to maintain current configuration control. This section documents acceptance by Plant Engineering, the System Manager, and Station Operations. This section includes: a Materials Disposition section providing disposition for those materials left over at the end of the modification; a Change Document List section which identifies documents impacted by the change that are not required for installation; an Exception List section which identifies work not completed; and a Close–out Check List which identifies actions to be completed prior to system restart and prior to change package close–out.

Walkdowns may be performed throughout the various stages of change package development and implementation using procedure NC.DE-AP.ZZ-0003(Q), "Modification Walkdown Program," (DEAP-3). Walkdowns are performed to verify existing configuration, determine causes of plant/document discrepancies, and determine the feasibility, operability, maintainability or testability of a proposed design change.

Change Package Review and Approval

The change package is reviewed through-out development, implementation, and close-out. Requests for design changes are reviewed by a review board. The

problem that the DCP is intended to correct, and the proposed design change, are presented to the review board by the System Manager and Design Engineer. The board then determines if the modification is necessary, and whether the proposed change will adequately address the problem.

Reviews from the technical disciplines, specialty areas, mandatory station department and the installer are documented on the Interface Record. The design change package requires approval from a PSE&G Engineering Supervisor, SORC review (for change packages requiring 10CFR50.59 Safety Evaluation), and station General Manager approval prior to implementation. The design change packages are peer reviewed and design verified if required. Documents required for system operation are revised following installation in accordance with the applicable controlled processes. The System Manager accepts the package for the station and confirms the system is ready for turnover to Operations. The Senior Nuclear Shift Supervisor (SNSS) then accepts the system for operability testing. Final close-out of the package occurs when impacted documents and databases have been updated or placed in a tracking system in accordance with their own controlled process.

The design change process interfaces with many processes and site organizations. The key supporting engineering processes include design bases/input, design calculations and analyses, Modification Walkdown Program, design drawings, peer review, Design Verification, Procurement Classification Guidelines, modification concerns and resolution, and engineering evaluation. Higher tier processes include the Document Management Program, SORC, Work Control Process, Preventive Maintenance Program, Corrective Action Program, Safety Tagging Program, Code Job Package, nuclear licensing and reporting, chemical control, and 10CFR50.59 applicability reviews and safety evaluations. The Change Process requires support from Operations, Maintenance, Planning, Radiation Protection, Chemistry, Licensing, Fuels, Quality Assurance, Engineering, Business Support, and Nuclear Training.

Training on the NAP-8 change process is required as part of the Engineering Support Personnel (ESP) Training Job Qualification Guide (JQG) for PSE&G engineers. Staff augmentation contractors and contractors who perform work off-site also are trained either on-site, or as a part of their contract if they are to perform design or design change work.

The NAP-8 procedure receives a biennial review as required by NAP-1. The review includes confirmation of the technical accuracy of the procedure, level of information provided by the procedure for its intended use, legibility of procedure pages, compliance with the procedural hierarchy requirements, and compatibility

of the procedure with referenced procedures and processes. The procedure may also be revised in response to conditions adverse to quality identified in accordance with the Corrective Action Program. Revision requests may also be generated by any procedure user. NAP-8 revisions require screening in accordance with the 10CFR50.59 program.

B. Minor Modification Process (NC.NA-AP.ZZ-0017(Q))

Procedure NC.NA-AP.ZZ-0017(Q), "Minor Modification Process," (NAP-17) describes the process for preparing, reviewing and installing minor modifications at both Hope Creek and Salem Stations. A Minor Modification is a change that does not impact the safe operation of the station, affect the design bases of any SSC, require a change to the TS, or create an USQ.

The Minor Modification process may be used provided the following criteria are met:

1. The modification is a result of a deficient condition in the field or is needed to enhance the reliability of performance of an SSC, and
2. The change does not impact the design bases information described in the UFSAR or other design bases documents, and
3. The change requires no complex calculations or engineering design, and
4. Complicated or specialty engineering analysis or unique equipment qualification is not required, and
5. The change can be implemented with minor engineering direction.

If a minor modification is appropriate, an engineer is assigned to develop a Minor Modification Package (MMP). The MMP addresses the purpose, description, quality classification, drawings affected, and special instructions or conditions for installation and return to service. The MMP contains a Design Inputs Evaluation which documents review of the design inputs. Items considered include consistency of the minor modification with design parameters, environmental and seismic qualification, electrical system loading, ASME Code requirements, performance characteristics of safety-related SSCs, effect on safety actuation systems, separation criteria and common mode failure, fire hazards analysis, personnel injury or equipment damage, and radiological concerns. Specialty reviews are performed as necessary in accordance with DEAP-7.

A 10CFR50.59 applicability review is also performed for minor modifications. If the minor modification is within the scope of 10CFR50.59, a safety evaluation is performed in accordance with NAP-59.

An independent design verification review is performed if the minor modification affects Q-listed SSCs, non-safety-related SSCs considered important to safety, or if required by management. The MMP is reviewed by an engineering supervisor, and if important to safety, by station Quality Assurance. If the minor modification is within the scope of 10CFR50.59, it is presented to SORC for review and recommendation for approvai by the station General Manager.

C. Temporary Modification Process (NC.NA-AP.ZZ-0013(Q))

A Temporary Modification (T-Mod) is defined as a modification to an operable SSC that temporarily alters the approved designed configuration. This includes lifted leads, jumpers, and electrical and mechanical installations. Procedure NC.NA-AP.ZZ-0013(Q), "Control of Temporary Modifications," (NAP-13) is common to both the Hope Creek and Salem Stations. This procedure ensures that T-Mods are controlled in a manner that ensures operator awareness, conformance with design intent and operability requirements, and preservation of plant safety and reliability. Proper use of the NAP-13 process ensures that the design process for permanent plant modifications will not be circumvented.

It is intended that temporary changes be minor in scope, of short duration and few in number. Activities not considered to be T-Mods are defined in NAP-13. Examples of activities not considered to be T-Mods include installation of a pressure gauge on an instrument tap, installation of a hose on a station air/breathing air system hose station, connection of sample tubing, and connection of hoses to service outlets specifically designed for that purpose. Under conditions defined in NAP-13, temporary modifications may be made without a Temporary Modification Package (TMP). These temporary modifications are controlled by approved procedures.

The need for a TMP is reviewed by a System Engineering or Nuclear Engineering representative. If a TMP is required, an engineering representative is assigned to develop a TMP which includes the following information:

1. Review and approval signatures.

2. Description of the T-Mod, including purpose, SSC affected, locations, quality classification, 10CFR50.59 evaluation, affected drawings and

procedures, special instructions for installation and removal, action required to remove, and expected removal date.

3. Design input evaluation, which addresses design characteristics such as pressure, temperature, fluid, chemistry, voltage, current, material compatibility, seismic, wind, thermal, and dynamic loading, environmental or seismic qualification, electrical system loading, effects on ASME Code Class components, effects on safety-related SSCs, separation concerns and common mode failure potential. Fire hazards analysis, potential personnel injury and equipment damage resulting from the T-Mod, effect on the operators ability to control or monitor the plant and radiological effects are also considered.

4. Record of installation and removal, including procedural controls to assure return to normal configuration and retest requirements.

A 10CFR50.59 Applicability Review is performed for T-Mods. If the T-Mod is within the scope of 10CFR50.59, a safety evaluation is performed and processed in accordance with NAP-59. This review process is completed prior to T-Mod implementation.

Following TMP development, a review is performed by a Nuclear Engineering Supervisor. SORC review is required if 10CFR50.59 is applicable, the T-Mod affects nuclear safety, or if requested by the engineering representative, supervisor, manager or SORC member. TMPs which require SORC review, also require approval by the station General Manager.

Installation of the T-Mod is authorized by the Senior Nuclear Shift Supervisor (SNSS) or Nuclear Shift Supervisor (NSS). The T-Mod is installed in accordance with the instructions contained in the TMP, and an independent installation verification is performed. The SNSS confirms that affected Operational Working Drawings have been updated upon installation of the T-Mod. The engineering representative verifies the applicable procedure revisions have been incorporated. Removal of the T-Mod requires SNSS/NSS authorization.

The Technical Document Room (TDR) notifies copy holders of operational working drawings affected by the installation and removal of T-Mods. The operating shifts review the status of installed T-Mods prior to changing modes and as part of shift routine. Installed T-Mods, if accessible, are reviewed quarterly to ensure that they are still required and that each is properly installed and the tags are correctly in place.

D. Nuclear Fuel Management Program (NC.NA-AP.ZZ-0072(Q))

Procedure NC.NA-AP.ZZ-0072(Q), "Nuclear Fuel Program," (NAP-72), defines an integrated fuel management program for the procurement, use, and disposal of nuclear fuel at Hope Creek. The Nuclear Fuels Section of the Nuclear Engineering Department is responsible for developing and maintaining the design bases for fuel and control blades, and ensuring that fuel designs loaded into the Hope Creek reactor cores and spent fuel pools conform to the design descriptions and constraints as described in the Hope Creek UFSARs and TS.

At the start of core reload design activities, design initialization discussions are held with the fuel vendors to address operational and design bases constraints that could impact core design. The Nuclear Fuels Section designs or oversees the core reload design to ensure that required analyses are completed and meet pertinent regulatory requirements and commitments. Fuel fabrication techniques, methods, processes and procedures are reviewed by PSE&G personnel to ensure design intent is met and that fabrication processes are appropriately controlled. Quality Assurance performs technical audits of the fuel vendors periodically, in accordance with this program. Reviews ensure that a Reload Analysis is performed using NRC approved computer codes, methods and procedures.

A 10CFR50.59 Safety Evaluation is performed for each core reload. In-house reviews are performed to validate the reload design analysis provided by the fuel vendors. For new fuel designs, additional operating margins are designed into the reload analysis. Operating experience with the new design is factored into determining the appropriate design margins to use in future cycles.

To ensure nuclear fuel design compliance, the Nuclear Fuels Group specifies required testing following core alterations, modifications, or instances when the core is not within analyzed design parameters. The technology is maintained to simulate the physical plant for safety, design, performance and transient thermal hydraulic evaluations. The Nuclear Fuels Engineers provide engineering support of station operations pertaining to nuclear fuel management, core performance and nuclear fuel performance.

IV. Description of Configuration Control Processes

PSE&G has the following processes in place to maintain plant configuration in accordance with the design bases. These processes are used in day-to-day station operations, maintenance and testing to control plant configuration, and interface with the design and configuration change processes described in Section III above.

A. Nuclear Procedure System (NC.NA-AP.ZZ-0001(Q))

Procedures govern the conduct of Station Operations, Department Operations, Corrective and Preventive Maintenance, Design and Configuration Changes, Testing, Procurement, Engineering Activities, Corrective Action Program and Oversight Activities. Procedures used at the Hope Creek Station are available to station personnel electronically via the Document Management System (DMS). Working procedures may be reproduced from the DMS for use by workers.

The requirements related to development, revision and use of procedures are contained in NAP-1, which is common to the Hope Creek and Salem Stations. Each procedure has a sponsor organization responsible for ensuring that new procedures and revisions receive the appropriate levels of review and approval. New procedures and revisions are subject to the 10CFR50.59 process as described in NAP-59. The 10CFR50.59 review process is the means by which procedure changes are evaluated for impact on the Hope Creek Station UFSAR and TS.

Development and revision of upper tier Nuclear Administrative Procedures (NAPs) governing activities within the scope of Regulatory Guide 1.33 (February, 1978), are reviewed by SORC, as required by Section 6 of the TS. SORC is also responsible for reviewing new procedures and revisions that are within the scope of 10CFR50.59.

As defined in NAP-1, Station Qualified Reviewers (SQRs) perform a standardized independent technical review of defined categories of procedures. The SQR review focuses on the bases for the change, and is intended to provide reasonable assurance that the procedure is technically correct and does not compromise nuclear safety. The need for a cross-discipline review is determined during this review. SQRs are required to meet or exceed the requirements of Section 4.1 and 4.7 of ANSI/ANS 3.1-1981 for the discipline(s) in which they are qualified, must complete SQR training, and maintain an active status through biennial recertification subject to the approval of the SORC Chairman.

Any person on site may recommend procedure changes by submitting a revision request via the Action Request Process. In addition, a revision request may be initiated as a result of a corrective action for a condition adverse to quality. Procedure revisions resulting from changes to the TS are identified as part of the license amendment process. Revisions or new procedures resulting from design change packages are initiated and controlled as defined in NAP-8.

The need for Verification and Validation (V&V) of new procedures and revisions is determined by the procedure sponsor organization in accordance with procedure NC.NA-WG.ZZ-0001(Q), "Procedure Writer's Guide." The V&V is performed in as near-to-real circumstances as possible, without actually manipulating installed plant equipment. During the V&V, the criteria applied include adequacy of meeting the intent of the procedure, conformance of the procedure with plant design, and satisfaction of commitments.

Administrative hold is used to place a procedure in the inactive status when it is determined that the procedure does not comply with station, regulatory, or licensing commitments. If an administrative hold is the result of a condition adverse to quality, the condition is addressed in accordance with the Corrective Action Program.

Biennial reviews are performed by the procedure sponsor organization for active and current procedures subject to the Quality Assurance Program in accordance with NAP-1. In defined cases, normal use of the entire procedure within the biennial review period may satisfy the biennial review requirement.

Training requirements for new procedures and revisions are determined by the procedure sponsor organization. The sponsor organization receives input regarding training requirements from procedure reviewers.

B. Managed Maintenance Information System (MMIS) (ND.DE-TS.ZZ-5409(Q))

The Managed Maintenance Information System (MMIS) is an important tool to assist site personnel in configuration control activities. Several databases reside in the MMIS system related to configuration control, including design and configuration change tracking, resource data, maintenance planning, and the Corrective Action Program (NAP-6). The system is used for both the Hope Creek and Salem Stations.

The MMIS contains component nameplate data, maintenance planning, inventory control, and purchasing information. The MMIS database is the basic tool to identify and maintain current information on SSCs, and identify SSCs that are subject to the QAP.

The MMIS Resource Data Module contains the necessary entry screens, inquiry screens, data files, and report definitions to enable site personnel to obtain data related to plant SSCs. Procedure NC.DE-AP.ZZ-0015(Q), "MMIS Resource Data

Module," (DEAP-15), defines the MMIS and provides directions for the control and use of the MMIS Resource Data Module data base.

Changes to data in the MMIS database are made through the change package process in accordance with NAP-8. Significant data are those data derived directly from design documents and include system, component identification, Bill of Materials (BOM) information, manufacturer identification and model number, and component quality classification. Procedures require that data base input be independently verified.

C. Work Control, Safety Tagging, and Post Maintenance Testing Programs

Procedure NC.NA-AP.ZZ-0009(Q), "Work Control Practices," (NAP-9), provides administrative controls for identifying, planning, scheduling, reviewing, performing, testing and post review of preventative and corrective maintenance. The work control process is designed to maintain configuration control during maintenance activities. It provides for proper planning of work activities, authorizations for removal of SSCs from service, controlled replacement of parts to prevent unauthorized design changes, post-maintenance testing to ensure equipment operability, and restoration of SSCs following maintenance activities.

Work Control (NC.NA-AP.ZZ-0009(Q))

Procedure NAP-9 describes the method for controlling work at the Hope Creek Station. Work at the station includes corrective maintenance, preventive maintenance, modifications, testing, experiments, inservice inspections, TS surveillance tests, refueling, and nondestructive examination activities.

Work activities are performed using procedures that specify how the activity is to be performed, plant conditions required, methods to be employed, equipment and materials to be used, and the sequence of the work activity. Work packages are developed for each maintenance activity, and are the official documentation of the work performed. The work package assures that work history is maintained for tracking and trending purposes.

The MMIS system is the primary tool used for the work control process. Work is identified, planned, scheduled and tracked on-line using the MMIS system. Component history data is stored in MMIS to provide access to records of previous work for planning and trending purposes. When MMIS is not available, manual work requests are used to identify and document work.

Work is identified using the Action Request process. Maintenance activities are prioritized using Hope Creek station specific criteria. These criteria consider whether the activity is corrective or preventative maintenance, the impact of the malfunction on safe plant operation, and TS Limiting Conditions for Operation (LCO) requirements.

Guidance is provided for planning maintenance in a manner that minimizes the potential for any compromise to plant safety. The planning and scheduling process considers the possible safety consequences of concurrent or sequential maintenance, testing or operations activities. The work package development process includes review of radiation protection, testing, and administrative requirements; required permits; spare parts; engineering support; and protective tagging requirements.

Safety Tagging Program (NC.NA-AP.ZZ-0015(Q))

Protective tagging requirements are determined using procedure NC.NA-AP.ZZ-0015(Q), "Safety Tagging Program," (NAP-15). This procedure controls and provides information regarding equipment status to protect personnel and equipment from unexpected energizing, startup or release of stored energy. This program is applicable to SSCs at Hope Creek. Equipment requiring safety tagging is identified based on the work to be performed, equipment operational restrictions, or procedural requirements. Personnel with site access receive specific training related to the safety tagging program.

Station Testing Program (NC.NA-AP.ZZ-0050(Q))

Post-maintenance testing requirements are determined by procedure NC.NA-AP.ZZ-0050(Q), "Station Testing Program," (NAP-50), applicable to both Salem and Hope Creek. Post maintenance testing verifies that the affected SSC is capable of performing its intended function, that the original malfunction, if applicable, has been corrected, and that no new or related problems have been created by the maintenance activity. Testing guidelines for electrical and controls, mechanical maintenance, valve maintenance, and Motor Operated Valves (MOVs) are provided in Attachments to NAP-50. Other sources of information for determining post maintenance testing requirements include TS Surveillance requirements, procedures, INPO Good Practice, the Inservice Inspection Program, the Inservice Testing Program, the Environmental Qualification Program, the Fire Protection Program, maintenance history, the UFSAR, and vendor manuals.

Work Package Review and Approval

In accordance with NAP-9, work packages are reviewed by the Work Control Center/Operating Shift prior to commencing the work activity. Considerations for this review include TS requirements, system interactions, safety tagging needs, possible changes in radiological conditions, and inadvertent Engineered Safety Feature (ESF) actuations. If needed, the tagging request is reviewed, approved and authorized for implementation. Qualified operators are briefed and any tagging requests are implemented prior to the start of removal of equipment from service.

Permission to commence the work activity is obtained from the SNSS or the NSS. Upon completion of the work activity, work is closed, and any tags are removed. When operability retesting is satisfactorily completed and reviewed by the NSS or SNSS, the equipment is returned to service. The work package is reviewed by the Job Supervisor and Planning Group to ensure applicable requirements were met.

D. Engineering Analysis

The plant processes governing engineering analyses include the design Calculation and Analyses process, the Dose Analysis, and the Engineering Evaluations. These processes are established to ensure that the design bases are considered during routine and modification related engineering activities.

Design Calculation and Analysis (NC.DE-AP.ZZ-0002(Q))

Procedure NC.DE-AP.ZZ-0002(Q), "Design Calculations and Analyses," (DEAP-2), establishes the technical and administrative requirements for development, maintenance, and control of design calculations in accordance with Regulatory Guide 1.64. This procedure is common to the Hope Creek and Salem Stations. The DEAP-2 procedure specifies that calculations include or reference:

1. Purpose or objective,
2. Design inputs or design bases,
3. Computer programs used as input,
4. Assumptions including those requiring future confirmation,
5. Conclusions, and

6. Actions necessary to support the conclusions.

DEAP-2 recommends that calculation preparers use the design bases/input procedure (DEAP-1) for design considerations and design input. The Probabilistic Risk Assessment (PRA) program is considered as a method to assess the potential risk impacts.

Dose Analysis

The Dose Analysis process is used to maintain dose analyses that are consistent with the design bases. Dose analyses are used to determine plant and offsite radiation dose rates, and integrated radiation doses, during normal operation and following design bases accidents. These analyses are performed for design and configuration changes, as necessary.

Dose analyses are documented in design calculations or vendor technical documents. Design calculations are controlled by DEAP-2 and are approved at the supervisor level. Vendor technical documents are controlled in accordance with the Vendor Technical Document Control Program, and are also approved at the supervisor level.

The Specialty Review process ensures that the effects of Change Packages or temporary modifications on dose analyses are evaluated. Dose analysis changes are reviewed for 10CFR50.59 applicability, and safety evaluations are performed when necessary, using the guidance provided in NAP-59. Conditions adverse to quality that involve dose analysis are dispositioned and corrected in accordance with the Corrective Action Program and the Action Request Process.

Engineering Evaluation (NC.DE-AP.ZZ-0026(Q))

Procedure NC.DE-AP.ZZ-0026(Q), "Engineering Evaluations," (DEAP-26), authorizes, defines and controls the use of engineering evaluations. This procedure applies to formalized evaluations performed to document reviews, analyses, conclusions, or recommendations, on topics such as root cause analysis, problem analysis, engineering alternatives, safety concerns, or economic considerations. The engineering evaluation process is common to the Hope Creek and Salem Stations.

Design considerations and design inputs to the engineering evaluation are identified using DEAP-1. Potential impact on specific engineering programs is also evaluated using the Specialty Review procedure. Evaluations that change the design bases for SSCs important to safety or the basis of analyses or

conclusions stated in the SAR, receive a 10CFR50.59 applicability review or safety evaluation, if required, in accordance with NAP-59. The preparer is responsible for identifying updates to documents impacted by the engineering evaluation.

Engineering evaluations are approved by the preparers functional supervisor, manager, and supervisors or managers responsible for commitments identified in the evaluation.

E. Design Drawings Process (NC.DE-AP.ZZ-0004(Q))

Design drawings are important in ensuring that plant configuration is maintained in accordance with the design bases because they depict or describe the designed SSCs as they actually exist. The design drawing control process, procedure NC.DE-AP.ZZ-0004(Q), "Design Drawings," (DEAP-4), provides direction for preparation, review and approval of new design drawings, and permanent revisions to design drawings. This procedure applies to both the Hope Creek and Salem Stations.

The Design Drawings procedure (DEAP-4) outlines the process for preparing, reviewing and approving design drawings. Existing drawings are revised, or new drawings are created, in order to reflect changes or additions. Before the drawings become part of the approved permanent design record, a change package is developed in accordance with the NAP-8 process. The change package may include an interim revision document. After the change package is implemented, the interim revision drawings are used to update the original permanent drawings to reflect the approved installed condition.

During drawing preparation activities, plant walkdowns and necessary calculations are performed. The walkdowns are performed in accordance with DEAP-3. Calculations are performed in accordance with DEAP-2. The new or revised drawing is then prepared and peer reviewed. Final drawing approval is by a PSE&G supervisor or his designee.

Design drawings are retained as part of the permanent design record and are microfilmed and stored in accordance with the document control and records management program. Drawings are available to site personnel electronically from the Document Management System (DMS).

F. Software Control (NC.DE-AP.ZZ-0052(Q))

Procedure (NC.NA-AP.ZZ-0064(Q), "Software and Micro-processor Based Systems (Digital Systems)," defines requirements for computer software control

for the Nuclear Business Unit. Procedure NC.DE-AP.ZZ-0052(Q), "Software Control," (DEAP-52), provides the controls applicable to critical software used by the Engineering Department to support or perform safety-related functions, or provide data upon which decisions can be solely made regarding plant operation, design or emergency response. This process is common to the Hope Creek and Sale. . ations.

The use of software to produce, or assist in producing calculations that are critical to the design of components important to safety can have a direct effect on safety functions in the plant. As a result, software used in these cases is subject to the same QAP controls and requirements as other facets of nuclear plant design, construction and operation.

DEAP-52 provides guidance for evaluation of existing software, development of new software, and revision of existing software. Guidance is also provided regarding procurement, validation and verification, turnover, maintenance, voiding and security of critical software. Critical software used to support a change package or temporary modification requires a review for 10CFR50.59 applicability in accordance with NAP-59. Computer errors identified in critical software are evaluated for applicability in accordance with the reporting requirements of 10CFR Part 21.

Procedure NC.DE-AP.ZZ-0054(Q), "Process Computer Maintenance and Modification Control Program," (DEAP-54), used by the Digital Systems Group, documents the method by which maintenance and modifications to existing process computer systems, and installations of new process computer systems, shall be performed. It describes the methods of test performance, how errors will be documented and corrected, and the degree of subsequent testing.

G. Procurement Process (NC.NA-AP.ZZ-0019(Q))

Procedure, NC.NA-AP.ZZ-0019(Q), "Procurement of Materials and Services," (NAP-19), details the requirements for procurement of materials and services, inventory control, and receipt of materials and equipment to meet the Quality Assurance Program requirements. The process is essential in the configuration control process because it ensures that replacement parts and new equipment installed in the plant meet appropriate technical, design bases, and quality requirements. This process is common to the Hope Creek and Salem Stations.

Procurement activities related to design bases maintenance and configuration control include those that ensure:

1. Design bases requirements are translated into purchase specifications,

2. Appropriate quality and technical reviews and approvals are obtained for procurements,

3. Technical and quality assurance requirements for spare parts and materials are met, and

4. Materials subject to the QA Program are evaluated upon receipt.

Requirements for materials and services are incorporated into procurement documents. These requirements are based on design bases considerations for the item or service involved. Procedure NC.DE-AP.ZZ-0013(Q), "Control of Purchased Material, Equipment and Services Program," (DEAP-13), defines a method for preparation, issuance and control of material, equipment, and Q-listed service specifications.

Procedure ND.QA-AP.ZZ-0015(Q), "Material Evaluation Nonconformances," specifies that incoming parts and materials are receipt inspected to ensure that the procurement document requirements are met. Material not conforming with applicable requirements is:

1. Clearly identified and segregated, if possible, to prevent inadvertent use,

2. Documented,

3. Evaluated for impact of the non-conformance, and reportability under 10CFR21, and

4. Reported to Licensing for reporting to the NRC, if required, under 10CFR Part 21.

Final disposition of the non-conformance can include repair, rework, use-as-is, return to vendor or scrap.

H. Vendor Technical Document Control Program (NC.DE-AP.ZZ-0006(Q))

Procedure NC.DE-AP.ZZ-0006(Q), "Vendor Technical Document Control Program," (DEAP-6) is designed to ensure proper review, control, and maintenance of vendor technical documents (VTDs) from the time of receipt through final disposition. The purpose is to ensure that only the proper and

current VTDs are available for the operation and maintenance of Hope Creek SSCs. Accurate vendor information is essential in ensuring that plant design bases are maintained during operations, maintenance and design change activities.

DEAP-6 applies to the processing and control of VTDs which relate to specific SSCs and is applicable at both Hope Creek and Salem Stations. Typical VTDs include drawings, calculations, technical manuals, seismic qualification documents, and Environmental Qualification (EQ) documents.

I. Technical Specification Surveillance Program (NC.NA-AP.ZZ-0012(Q))

Procedure NC.NA-AP.ZZ-0012(Q), "Technical Specification Surveillance Program," (NAP-12) establishes the requirements pertaining to the scheduling and implementation of TS surveillance requirements for the Hope Creek Station. The NAP-12 procedure provides direction for:

1. Maintaining the TS cross reference matrix, and the MMIS Recurring Tasks related to surveillance testing activities,

2. Scheduling, initiating and implementation of routine, non-routine, and compensatory surveillance testing activities,

3. Applying the TS 4.0.2 maximum allowable extension of 25% to the normal surveillance interval,

4. Review and approval of surveillance test results, and

5. Extending surveillance test completion beyond the TS due date.

Procedures which define programs that implement the TS surveillance program are prepared and processed in accordance with NAP-1. Hope Creek maintains a TS cross reference matrix that relates each TS surveillance requirement to the implementing document, responsible department, mode requirements, surveillance interval, MMIS recurring task numbers, TS condition or event for non-routine surveillance tests, and initiating document for non-routine surveillance tests. This cross reference is updated when changes are made to the program resulting from TS Amendments, new or revised implementing procedures, and other changes.

Routine surveillance requirements with intervals of greater than seven days are scheduled in MMIS. Recurring tasks associated with MMIS scheduled surveillance tests are maintained, scheduled, and performed in accordance with NAP-9. Changes to surveillance testing procedures are reviewed for 10CFR50.59 applicability, and safety evaluations are performed, if required, in accordance with NAP-59.

V. Document Control Process (NC.NA-AP.ZZ-0003(Q))

Procedure NC.NA-AP.ZZ-0003(Q), "Document Management Program," (NAP-3), which includes both Document Control and Records Management functions, includes controls to assure that required documents are controlled and retained. The program assures that quality documents, including procedures, vendor technical documents, design change packages, design drawings, calculations and other engineering documents, are reviewed and approved by authorized personnel prior to issuance and use, and are retained in accordance with regulatory requirements.

The Document Control System (DCS), Design/Configuration Change Management System (DCCMS) and Document Management System (DMS), identify the status, history and current revision of controlled documents. When changes are made to quality documents, users can verify pending or approved changes in these systems. The DMS includes an electronic system for retrieval of effective and superseded versions of procedures, drawings and documents. This permits ready verification of the applicable version of documents and tracking of changes made in each revision of a document.

Each site department coordinates with the Document Management Group to establish a Records Type List (RTL) for their department. The RTL includes a list of retention requirements for current and anticipated record types maintained by the Document Management Group. When documents listed in the RTL are deleted, superseded or completed and required data is provided, documented and authenticated, they are transferred to Records Management. A copy of each completed record is either scanned into the DMS in accordance with NRC Generic Letter 88-18, converted to a microfilm, or retained as a hard copy original.

The Document Control and Records Management Programs interface with several plant programs. These include the Nuclear Procedure System for review, approval and issuance of procedures, the Document Management Program for issuance, distribution and retention of quality documents, the SORC for review of upper tier procedures and review of quality documents when an

evaluation under 10CFR50.59 is required, the DCP procedure for review and approval of design changes, the Vendor Information and Processing Program for control of vendor documents, the 10CFR50.59 process for changes to the facility, test or experiments or changes to procedures, and the Software Control Process.

The overall responsibility for the Document Control and Records Management Program resides with PSE&G Nuclear Business Support. Department Managers are responsible for ensuring that documents are approved and records are transferred to the Document Management Group, in accordance with interfacing procedures.

VI. 10CFR50.59 Process (NC.NA-AP.ZZ-0059(Q))

10CFR50.59 provides a mechanism for licensees to change the design of the facility as described in the Final Safety Analysis Report (FSAR), revise procedures described in the FSAR, and conduct tests and experiments not described in the FSAR, without prior NRC approval. This regulation requires a determination and written documentation providing the bases for concluding that the change or test does not involve an USQ, or a change to the TS. If a USQ or change to the TS is involved, the activity must be modified so that an USQ or TS change does not exist, or NRC approval must be obtained prior to implementing the activity.

The 10CFR50.59 Applicability Reviews and Safety Evaluations Procedure (NAP-59) is common to both the Hope Creek and Salem Stations. This process applies to design and configuration changes, tests or experiments, FSAR changes, procedural changes, temporary modifications, changes to specific technical programs, and certain dispositions of degraded and non-conforming conditions as documented by the Corrective Action Program.

As directed by appropriate procedures, proposed activities are screened to determine if they are within the scope of 10CFR50.59 by performing an applicability review. An activity is considered to be a change to the facility, as described in the FSAR, if it affects the design bases, operation, or functions of SSCs as described in the FSAR. If the proposed activity involves an SSC not described in the FSAR that could affect the design or operation of SSCs described in the FSAR, an evaluation under 10CFR50.59 is performed. In determining applicability under 10CFR50.59, the preparer is required to review the proposed activity against the UFSAR, NRC Safety Evaluation Reports (SER) and TS. The applicability review is formally documented.

If the activity is determined to be within the scope of 10CFR50.59, a Safety Evaluation is performed. The purpose of the Safety Evaluation is to determine if the proposed activity involves an USQ or a TS change. Guidance for answering the required 10CFR50.59 questions is provided in a Nuclear Administrative Standard NC.NA-AS.ZZ-0059(Q), "10CFR50.59 Program Guidance," (NAS-59). The Safety Evaluation undergoes a peer review. It is then reviewed by the SORC and approved by the respective station General Manager prior to implementation. If the proposed activity involves a USQ or a TS change, NRC approval is obtained prior to implementing the activity. An independent review is performed by the Offsite Safety Review (OSR) staff to verify that the changes implemented under the 10CFR50.59 process did not constitute an USQ.

The responsibility for NAP-59 and required reports rests with the Nuclear Licensing Department. Each site organization performing applicability reviews or safety evaluations is responsible for implementation of the program requirements.

The 10CFR50.59 Program requires that only qualified personnel perform applicability reviews and safety evaluations. PSE&G has established uniform training and qualification requirements for preparers, peer reviewers, and approvers. Required training of preparers and reviewers is performed by the Nuclear Training Department.

The 10CFR50.59 process interfaces with site processes and programs that have the ability to change the facility and procedures as described in the UFSAR. Examples of these processes include the Procedure Control, Corrective Action Program, Design Change, Temporary Modification, Minor Modification, UFSAR and TS Basis Change, and Design Calculation and Analysis.

VII. Technical Specification Amendment Process (NC.LR-AP.ZZ-0008(Z))

Procedure NC.LR-AP.ZZ-0008(Z), "Operating License and Technical Specification Change Processes," (LRAP-8), describes Technical Specification License Change Request (LCR) development and submittal, approved amendment receipt and implementation, and changes to the TS Bases. The process is common to the Hope Creek and Salem Stations.

Any Site department may initiate an LCR by submitting a request per procedure NC.NA-AP.ZZ-0035(Q), "Nuclear Licensing and Reporting," (NAP-35). UFSAR and TS Basis changes related to the LCR are submitted in this manner as well.

The LCR is developed by the Licensing and Regulation Department staff using guidance contained in LRAP-8. Factors considered during LCR development include supporting evaluations and analyses, similar amendments developed by other utilities, generic guidance such as Improved Standard TS, the Standard Review Plan, the Code of Federal Regulations, the UFSAR and SERs, previous TS amendments, and Utility Owners Group information. The LCR is developed using a standard template and writers guide included in LRAP-8. Following review and concurrence by the affected departments, the LCR is reviewed by SORC and the Offsite Safety Review committee. The LCR is approved by a company officer prior to submittal to the NRC.

An Implementation Plan is normally developed by Licensing and affected site departments after LCR development and just prior to submittal to the NRC. The Implementation Plan considers items including procedure changes, UFSAR changes, design changes, setpoint changes and training. Tracking items are assigned for each identified action. The Implementation Plan is finalized when the approved License Amendment is received. The License Amendment is issued following completion of necessary implementing actions and the approved TS amendment is given to Document Control for distribution. The TS, TS Bases and NRC amendment Safety Evaluation are available to site personnel on the DMS system.

VIII. 10CFR50.71(e) UFSAR Update Process (NC.LR-AP.ZZ-0013(Z))

10CFR50.71(e) requires that the UFSAR be revised to include the effects of changes made to the facility or procedures as described in the UFSAR, safety evaluations performed by the licensee either in support of license amendments or in support of conclusions that changes did not involve an unreviewed safety question, and analysis of new safety issues performed by or on the behalf of the licensee at Commission request, since the prior revision of the UFSAR. Changes to the UFSAR are reported to the NRC either annually or 6 months after each refueling outage, not to exceed 24 months, in accordance with 10CFR50.71.

Procedure NC.LR-AP.ZZ-0013(Z), "UFSAR Maintenance Process," (LRAP-13), maintains the accuracy of the Hope Creek UFSAR through regular revisions to incorporate changes identified through the design change process, the Corrective Action Program, and other internal processes and reviews. NAP-59 specifically requires that the UFSAR be reviewed and the Safety Evaluation Form provide a space for documenting any UFSAR Change Notices generated.

Conditions adverse to quality affecting the UFSAR are documented and dispositioned in accordance with the Corrective Action Program. Design and configuration changes are reviewed for effects on the UFSAR. A change notice is prepared for required UFSAR changes and is included as a "change document" in the Change Package.

The SORC approval dates are used for tracking the update requirements contained in 10CFR50.71(e). Changes to the plant are reflected in the UFSAR following installation so that the UFSAR reflects the plant "as-built" condition.

The UFSAR is provided to plant personnel electronically on the DMS computer. Controlled copy distribution of the UFSAR and change notice lists, and scanning of the UFSAR and change notice lists onto DMS, is performed through the Document Management Program.

The responsibility for UFSAR maintenance rests with the Licensing and Regulation Department which manages the process, develops UFSAR revisions, and approves UFSAR change notices. The Engineering Department is responsible for maintaining an accurate description of systems, structures and components in the UFSAR through the preparation of UFSAR change notices. The SORC reviews and approves safety evaluations related to UFSAR change notices. The OSR Group reviews related safety evaluations for unreviewed safety questions. The Document Management Group performs controlled copy distribution of the UFSAR and change notice lists, and scans these into DMS. Site organizations are responsible for maintaining the UFSAR description for their organization, processes and procedures, and preparation and review of UFSAR change notices related to the organization's area of responsibility.

UFSAR Change Notices are tracked using a DMS database. Approved change notice lists and copies of the UFSAR mark-up books are provided to station personnel via the DMS system. The station SNSS has access to hard copies of the UFSAR mark-up books, and approved change notice lists for pending UFSAR changes.

IX. Specific Technical Programs

Procedure ND.DE-AP.ZZ-0038(Q), "Engineering Programmatic and Technical Standards," (DEAP-38), establishes requirements for preparation, identification, review and approval, issuance and maintenance of Programmatic and Technical Standards. A Programmatic Standard is a design document developed for a specialized area where a history of licensing compliance needs to be developed and maintained.

The following programs are examples of specialized areas that are governed by Programmatic Standards:

A. Equipment Environmental Qualification Program (DE-PS.ZZ-0002(Q))

The Hope Creek Environmental Qualification (EQ) Program provides assurance that electrical equipment important to safety located in harsh environments will perform its safety function when called upon to do so during normal, abnormal and accident conditions. Equipment important to safety includes safety-related Class 1E electrical equipment, non safety-related electrical equipment whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of required safety functions by safety-related equipment, and certain post-accident monitoring equipment.

A description of the EQ program is outlined in UFSAR Section 3.11. In addition, the EQ Programmatic Standard DE-PS.ZZ-0002(Q), "Environmental Equipment Qualification Program," (DEPS-2), and Administrative Procedure NC.NA-AP.ZZ-0062(Q), "Environmental Qualification Program," (NAP-62), provide information on the implementation and interface process for the EQ Program.

The Hope Creek EQ Program meets the intent of NRC requirements of 10CFR50.49 and assures the qualification is consistent with NUREG 0588 Category I guidelines and Hope Creek's licensing commitments.

Administrative changes to the NAP-62 program are performed in accordance with NAP-1. Reviews and approvals are required of the EQ Program sponsors, SORC and the station General Manager-Operations. Design Change Packages affecting the EQ Program are subject to the Specialty Review Process.

The EQ specialty review checklist assures that changes to the EQ Program are captured. Within the EQ Program, programmatic controls including Equipment Qualification Maintenance and Surveillance Instruction Sheets (EQMSIS) provide maintenance requirements for the EQ equipment. Environmental Qualification related maintenance or inspection tasks are uniquely identified in the MMIS system to ensure that the task is implemented by the due date. Deferral of maintenance tasks for EQ equipment is not permitted without prior evaluation from the EQ Engineer. The EQ Maintenance Engineer performs an independent verification to ensure that the EQ preventive maintenance actions are implemented.

B. Motor Operated Valve Program (NC.DE-PS.ZZ-0033(Q))

This program applies to the population of Hope Creek MOVs that perform a specific fluid system design bases function to assure that concerns of NRC Generic Letter 89-10 are appropriately addressed. The purpose of this program is to verify MOV operability under operating, abnormal or emergency operating design bases conditions to assure that Generic Letter 89-10 issues are appropriately addressed. The program also addresses the maintenance of MOV data and calculations to assure continued MOV operability for the life of the plant.

The MOVs included in the scope of the program are determined in accordance with the screening criteria contained in the MOV Program Programmatic Standard NC.DE-PS.ZZ-0033(Q), "Motor Operated Valve Program," (DEPS-33). For each valve determined to be within the scope of the program, an operating conditions evaluation is performed. This evaluation determines the appropriate design bases parameters present during valve operation under the worst case operating conditions. Typical documents reviewed to determine the valve operating design bases include the UFSAR, TS, CBDs, normal, abnormal and emergency operating procedures, plant drawings, design calculations and component test data and vendor information. An electrical and mechanical capability review is then performed and available diagnostic test data is reviewed. The results of these reviews are compiled into an MOV Capability Assessment which is a summary of the conclusions developed from the design bases review, and documentation of the assumed values, calculations and limitations for each MOV.

DEPS-33 establishes the general guidelines for programmatic valves. Exclusion of valves from the MOV Program is accomplished by use of the change package peer review process, and an independent valve specialist review for concurrence. Design verification of changes, and calculation verifications are performed using the Design Engineering procedures.

The preventative maintenance repetitive tasks process includes valve testing and maintenance. The Corrective Action Program documents and evaluates conditions adverse to quality relative to the MOV Program. Corrective Maintenance is identified through the Action Request process.

The engineering and maintenance organizations have responsibility for implementation of the MOV Program. Responsibilities for the design bases, programmatic development and evaluations reside within the Specialty Engineering Branch of Systems Engineering.

C. Fire Protection Program (NC.DE-PS.ZZ-0001(Q))

The Fire Protection Program as outlined in the Hope Creek UFSAR and in procedure NC.DE-PS.ZZ-0001(Q), "Fire Protection," (DEPS-1), provides a program to identify and control design features unique to fire protection which forms an integral part of the PSE&G Fire Protection Program. The Fire Protection Program includes the identification of fire protection requirements for safe shutdown equipment and systems, fire detection and suppression systems, fire resistive structures, administrative controls and procedures to assure equipment reliability and to minimize fire hazards, training of personnel, and maintenance of a dedicated fire brigade. The Fire Protection Program has been unified to serve both Hope Creek and Salem. Although both stations have different license and design bases commitments, the fire protection program is designed to address and accommodate both plants.

The programmatic standards, design change process, and the document configuration control processes assure that the fire protection design features are maintained. The level of fire protection review for design changes not involving a fire protection feature is performed in accordance with DEAP-7. If any items on the Fire Protection Specialty Review Check Sheet are identified by the Project Team Member or a Peer Reviewer as impacting fire protection the design change is reviewed to determine any impact on the Fire Protection Program.

A detailed evaluation is performed for activities that could potentially impact the fire protection program. The Fire Protection Program assures compliance with NRC Branch Technical Position CMEB 9.5-1.

In accordance with NAP-1, the level of review for fire protection procedure changes includes an originator from the Loss Prevention Group, (formerly the Site Fire Protection Group) with department review, and a department manager's approval. Changes to fire protection procedures are also subject to a 10CFR50.59 applicability review. If a safety evaluation is performed, SORC review and the Station General Manager's approval are also required.

D. Inservice Testing & Inservice Inspection Programs (NC.NA-AP.ZZ-0027(Q) and NC.NA-AP.ZZ-0070(Q))

Procedures NC.NA-AP.ZZ-0070(Q), "Inservice Testing," and NC.NA-AP.ZZ-0075(Q), "Valve Programs," apply to Inservice Testing (IST) of Nuclear Class 1, 2 and 3 pumps and valves required to perform a specific function in shutting down the reactor to the cold shutdown condition or in mitigating the

consequences of an accident. The Inservice Inspection (ISI) Program control procedure NC.NA-AP.ZZ-0027(Q), "Inservice Inspection Program," applies to Nuclear Class 1, 2 and 3 pressure retaining components and their supports. The requirements for these programs are defined in 10CFR50.55a(g) and the Hope Creek UFSAR.

The IST and ISI Programs are subject to PSE&G management review and reviews by the Authorized Nuclear Inspector. The IST and ISI Programs interface with the Design Control Process, the 10CFR50.59 Process, the Work Control Process, and the Corrective Action Process. The specialty review process requires that modifications proposed to Nuclear Class 1, 2 and 3 components and their supports will be processed through the Specialty Engineering (ISI or IST Group, as applicable). Also current regulations require updating of the ISI and IST Programs once every ten year interval in accordance with 10CFR50.55a.

The IST group develops and maintains the IST Program. Tests required by the program are performed by the Operations Department, Maintenance Department and the Inservice Inspection and Test Group. Data evaluation is performed as required by the IST program, and results are trended.

The ISI group develops, maintains and implements the required examinations in accordance with the ISI Program Long Term Plan. To maintain the ISI Program accurate and updated, plant design changes are reviewed for ISI applicability in accordance with the specialty review process. To maintain control of repairs and replacements under ASME Section XI, Code Job Packages and their revisions are routed to Specialty Engineering (ISI Group) for review and acceptance. The Code Job Packages control applicable post maintenance testing and preservice Inspections.

Paragraph 10CFR50.55a(3) provides guidance for submitting proposed alternatives (relief requests) to the ASME Code. The need for an ASME Code relief request is identified by the IST or ISI Group and developed and processed by the Licensing and Regulation Department. Typically, the relief request consists of a cover letter and attachment(s) which demonstrate that proposed alternative testing or examination would provide an acceptable level of quality and safety, or compliance with the specified requirement of the ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. NRC approval of the relief request must be obtained prior to implementation.

X. Independent Oversight

A. Station Operations Review Committee (NC.NA-AP.ZZ-0004(Q))

The Station Operations Review Committee (SORC) advises the station General Manager on matters related to nuclear safety. As defined in procedure NC.NA-AP.ZZ-0004(Q), "Station Operations Review Committee," (NAP-4). SORC reviews plant operations, modifications, procedures, tests, temporary modifications, 10CFR50.59 Safety Evaluations, unit trip reports, and other issues as delineated in TS 6.5.1. The TS review requirements are reflected in NAP-4. This procedure is common for both Salem and Hope Creek.

SORC reviews specifically defined issues regarding changes to the design bases, licensing bases, and accident analyses in Chapter 15 of the UFSAR. In reviewing safety evaluations required by 10CFR50.59, SORC determines if the conclusion whether an USQ is involved was properly made. The SORC review also considers the radiological safety effect on personnel and any effect on safety related systems/components the change may cause. SORC ensures that the preparer of the change has outlined the relationship of the proposal to both safety related and important to safety systems, structures and components, and issues raised have been addressed. Issues identified by SORC are documented and tracked by the Corrective Action Process.

SORC is comprised of members from station staffs who act as members and alternates as approved by the General Manager. They must meet the minimum qualifications listed in ANSI/ANS 3.1-1981. The membership of SORC is identified by position title in the Technical Specifications. The qualifications of each individual comprising SORC require documentation. Also, management expectations have been established for SORC members and presenters.

B. Off-Site Safety Review (ND.SN-AP.ZZ-0001(Q))

The Off-Site Safety Review (OSR) staff performs independent review activities consistent with the requirements of TS in the areas of Operations, Engineering, Chemistry, Metallurgy, Instrumentation and Controls, Radiological Safety, Quality Assurance, Non-destructive Testing, and Emergency Preparedness. Documented reviews are performed in accordance with procedure ND.SN-AP.ZZ-0001(Q), "Independent Safety Review Program." Areas reviewed include 10CFR50.59 safety evaluations, proposed changes to procedures or equipment and tests/experiments that involve an USQ, proposed changes to the facility Operating License and the TS, NRC violation responses and events warranting a review to verify that root causes and safety significance have been determined, and that proposed corrective actions appear effective and have or will be taken.

This procedure is common for both Salem and Hope Creek. The results of reviews are reported to the SORC.

C. On-Site Safety Review (ND.SN-AP.ZZ-0001(Q))

The On-Site Safety Review Group (SRG) staff performs reviews in accordance with procedure ND.SN-AP.ZZ-0001(Q), "Independent Safety Review Program," of selected plant operating characteristics, NRC requirements and guidance, industry advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. Also reviewed are selected facility features, equipment, and systems, as well as procedures and plant activities including maintenance, modifications, operational problems, and operational analysis. This procedure is common for both Salem and Hope Creek. The SRG staff members function as voting members of the SORC.

D. Quality Assurance

Quality Assurance (QA) performs planned and periodic audits and surveillances of the ongoing design change and configuration process. QA also samples design and configuration changes to ensure Quality Assurance requirements such as inspection and test requirements, acceptance requirements, and test results documentation are incorporated into the changes. Specifications for Q-listed materials, equipment and services are reviewed and approved by QA.

XI. Other Processes With the Potential to Affect Configuration Control

In addition to the processes discussed above, there are other operational practices by which traditional configuration control methods might be circumvented; however, appropriate procedural controls are applied to assure that the design bases is adequately considered and protected. These practices include such activities as Technical Specification Interpretations, Operator Aids, and procedures for removing and restoring systems to service following maintenance.

A. Technical Specification Interpretations (HC.OP-AP.ZZ-0111(Q))

In cases where TS requirements are considered ambiguous, formally documented interpretations may be generated to provide management clarification of the intent. Procedure HC.OP-AP.ZZ-0111(Q), "Technical Specification Interpretations," defines controls for use at Hope Creek.

When generated, these interpretations are reviewed and approved for consistency with the design bases. The interpretations are documented, reviewed for applicability of 10CFR50.59, reviewed by SORC, and approved by station management. In addition the interpretations are periodically reviewed for continued applicability and need. If appropriate, the content of interpretations may be incorporated into operating procedures. Technical Specification interpretations do not result in changes to the design bases for the facility.

B. Department Night Orders (NC.NA-AP.ZZ-0005(Q))

Night Orders as defined by procedure NC.NA-AP.ZZ-0005(Q), "Station Operating Practices," are a mechanism for issuing management instructions to Operations Department personnel. Such instructions may encompass daily schedule matters, special events or activities, housekeeping matters, non-routine plotting of process parameters, notices about personnel actions, requirements to read certain publications or documents, or similar matters. The procedure specifies that Night Orders should neither conflict with existing procedures, nor should they be used in place of procedures.

C. Control of Operator Aids (NC.NA-AP.ZZ-0044(Q))

Operator Aids as defined by procedure NC.NA-AP.ZZ-0044(Q), "Station Aids and Labels," are a category of locally mounted information, including sketches, notes, graphs, instructions, drawings or other documents. Aids are used to provide warnings to plant personnel performing work or testing in accordance with approved procedures, or to provide information, e.g., component name, or number to assist station personnel. They are not intended to be used to provide instructions for maintenance or testing. Operator Aids are controlled by each operating department for aids applicable to their department. Each department maintains a log of each Aid initiated by that department, controls the posting of the Aid, periodically reviews the continued need for the Aid, and assures that it is removed when no longer necessary. Operator Aids are not used to change the configuration or affect the plant design bases.

D. On-the-Spot Changes (NC.NA-AP.ZZ-0001(Q))

On-the-spot changes may be made to previously approved procedures, as authorized by procedure NAP-1, to permit completion of the activity prescribed by the procedure as long as the intent of the procedure is not changed, and the change does not alter a commitment. These changes typically involve correction of editorial or typographical errors identified during procedure execution. Procedure NAP-1 requires that the procedure change be documented and reviewed by the job supervisor or department management and approved by the

SNSS or NSS before performance. Subsequent review and approval of on-the-spot changes is required to assure 10CFR 50.59 review and to provide for permanent incorporation into the procedure. Copies of procedures including on-the-spot changes are incorporated into the document control system pending document revision.

E. Equipment Control Program (HC.OP-AP.ZZ-0108(Q))

Operability of SSCs is determined and tracked by the Operations department for Hope Creek by procedure HC.OP-AP.ZZ-0108(Q), "Operability Assessment and Equipment Control Program." Equipment declared inoperable is evaluated against the action statements of the TS, and the design bases for the SSC, and necessary action taken. If appropriate, an Action Request is processed for repair, modification, or replacement of the inoperable equipment. Inoperable equipment is formally tracked along with the necessary compensatory actions prescribed by the TS action statements.

XII. Availability of Design bases Documentation

In Section V. above, the Document Control Process is described. This process assures that documentation is available for use in evaluating potential changes to the facilities design bases. Procedure NAP-3 contains an overview of the document management system for the NBU. Record systems which PSE&G utilizes for retention of records include provision for easy retrieval. The primary systems for record storage and retrieval include the Document Management System, Microfilm Record Files, and the Managed Maintenance Information System.

Document Management System (DMS): The DMS is a computerized system for retention and retrieval of current configuration documents, including drawings, licensing manuals such as Technical Specifications and UFSAR, vendor documents, engineering documents such as calculations, drawings and design changes (change documents and modification documents); Configuration Baseline Documentation manuals; administrative and operating procedures; correspondence; and other similar classes of documentation. These classes of documentation must be widely available, and controlled since several classes of these records are subject to revision. In addition to current configuration documentation, limited historical documentation is also retained for reference in this system. Access terminals and printers are strategically located in the plant and in the Nuclear Administrative Building (NAB), permitting easy verification of revision status, and retrieval of working documents.

Microfilm Records: Historical documentation, including design change records, maintenance work orders, documentation furnished by Architect Engineers and Vendors, etc., is filmed and made available from microfilm libraries located throughout the NBU, including the Technical Document Rooms for both Salem and Hope Creek, and the NAB. Libraries are equipped with film readers with reproduction capability. The microfilm libraries include records which originated before the DMS system was fully operational.

Managed Maintenance Information System (MMIS): A computerized system is used on the central mainframe computer to control, track, and permit retrieval of equipment configuration information, maintenance work orders, Action Requests (ARs), and recurring maintenance work order tasks. The Document Control System portion of MMIS is also used to verify the current revision and revision status of documents. The MMIS system is accessible through the facility computer network through use of individual personal computers available to plant staff who have access needs.

To assure ready availability of these records to plant personnel, PSE&G has provided Technical Document Rooms, and a record system for Licensing manuals and correspondence.

Technical Document Rooms: Salem, Hope Creek, and the NAB have reference rooms which retain copies of documentation related to ongoing activities, a microfilm library, and copies of applicable reference manuals.

Licensing Files: Copies of correspondence related to requests for license change, responses to NRC reports, and other submissions to the NRC are retained in the NAB, and in a computerized Licensing Data Base, which is available for access through the full text search option of the local area computer network. The full text search has the capability to search a large computer based file of documents for specified words or phrases. A library of correspondence between PSE&G and regulatory agencies, along with reference materials is also maintained in the NAB.

The above systems assure the availability of reference documentation necessary for those who prepare, review or approve design change packages, minor modification packages, procedures, and other processes that could impact the design bases for the Hope Creek station. These processes are also available for those personnel with operational, maintenance, planning, and licensing responsibilities.

During two external reviews in 1995 conducted by Ogden Environmental and Energy Services Company, it was concluded that "the Design bases was readily retrievable (largely due to the DMS system, which facilitates easy access to design documents)," and "Document control and retrieval was excellent."

XIII. Personnel Performance Improvement

The above sections have discussed the various NBU programmatic and procedural systems and controls which contribute to maintenance of the design bases. This section discusses initiatives taken by PSE&G to provide training and documented qualifications for those staff who utilize these systems. The training has the potential to improve awareness and decrease the potential for compromising the integrity of the design bases. An increase in the level of awareness of personnel performance, as it relates to the design bases has been initiated by a strong policy statement . These NBU expectations were expressed by the Senior Vice President--Nuclear Operations regarding conformance with the design bases. Additionally, the training and qualification programs for engineering support personnel have been upgraded.

On July 25, 1996, the Senior Vice President--Nuclear Operations, issued to licensed operators, with distribution to other managers in the NBU, a letter entitled "Expectations for the Knowledge Level of the Licensing and Design Bases." The purpose of that memorandum was to communicate his expectations regarding understanding and applying licensing and design bases information during the performance of licensed activities.

Also, procedure NC.NP-PO.ZZ-0012(Q), "Training, Qualification and, Certification," (NPPO-12), was revised to reflect the new cultural behavior. This procedure provides guidance regarding the performance of training, qualification activities, documentation of training, qualification, and certification to satisfy regulatory requirements, industry standards, job requirements and other competencies. The implementing procedures apply to the NBU organizations.

One of these implementing procedures, NC.TQ-TC.ZZ-0905(Z), "Engineering Support Personnel Training Program Description," (TP-0905), provides training requirements for initial and continuing training for Engineering Support Personnel (ESP). The program is designed to train PSE&G engineers, engineering supervisors, and staff augmentation contractors. These include: Reactor Engineers, System and Maintenance Engineers, In-service Inspection and In-service Testing Engineers, Design Engineers, Plant Engineers, Licensing Engineers, Nuclear Safety Review Engineers, Nuclear Fuel Engineers, Procurement Engineers, Digital Systems Engineers, and Quality Assurance

Engineers. An additional training module has been added to those defined in TP-0905 entitled "Design bases Fundamentals." This training is currently being given to ESP in the above categories. The scope of the course includes definition of the plant design bases, identification of documents which constitute the design bases, demonstration of techniques to access design bases documents, discussion of regulatory requirements and examples of previous failures to control design configuration.

The Nuclear Design Engineering Department has also recently initiated a Design Engineering Review Board to evaluate the ongoing competency of engineers. This process utilizes selected Licensing and Engineering managers and supervisors, sitting as a review board, to interview and orally examine the knowledge and competence of individual engineers. The oral examination includes questioning on various aspects of the design bases, assessing design bases information in support of the answers to the questions, and the procedures which are designed to assure their maintenance.

Continuing maintenance of the design bases for the station is expected to result from effective implementation of these personnel training and qualification programs, coupled with the existing procedural and programmatic programs.

XIV. Summary

The above administrative and procedural controls are intended to prevent unreviewed or unauthorized alteration of the design bases for the plant. These controls provide reasonable assurance that facility operations are consistent with the TS and their bases, as well as the design bases described in the UFSAR and other design documents. Provisions have been defined to permit emergency changes to these procedures and controls when operational conditions require, or errors are discovered; however, procedures for review, approval and implementation include controls to assure that these changes are identified, reviewed and subsequently incorporated into the approved design packages, procedures, and training programs.

PSE&G is confident that these procedures are adequate in scope to address 10CFR50, Appendix B, 10CFR50.59 and 10CFR50.71(e), and they provide adequate control over the design and configuration. Should implementation problems or deficiencies in the program be detected in the future, they will be reported to the Commission as warranted in accord with 10CFR50.72 or 50.73. Appropriate corrective actions will also be taken.

RESPONSE TO QUESTION B

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

I. Overview

The purpose of this section is to provide the rationale for our reasonable assurance that the design bases requirements are appropriately translated into the operating, maintenance, and testing procedures. The primary engineering design and configuration control processes are discussed in "Response to Question A". Key attributes of these processes applicable to the translation of design bases requirements into operating, maintenance, and testing procedures are as follows:

- implementing procedures governing operations, maintenance, and testing activities are controlled and readily available for plant uses
- implementing procedures (and related documents) accurately reflect design requirements
- overarching processes (e.g., processes that control changes to design requirements, plant design, maintenance and operations activities) are defined and appropriately implemented

The fidelity of the Hope Creek design bases was described above under the heading of "Adequacy of Design Bases." As discussed, there is reasonable assurance that the design bases were adequately developed and maintained over time. This section will concentrate on the translation of the design bases into plant procedures. Where appropriate, the discussions identify known weaknesses and detail actions taken and/or planned to resolve these matters.

The "Process Controls" subsection of this response discusses key elements of the processes described in "Response to Question A" as they affect the relationship between the design bases and operations, maintenance, and testing procedures. The "Validation Reviews" subsection of this response describes actions which provide reasonable assurance that procedures are consistent with the plant design bases and identify potential process issues or improvements, where appropriate. The "Assessment of Effectiveness" subsection provides an assessment of the success with which the process controls provide reasonable

assurance for proper translation of the design bases into operating, maintenance, and testing procedures. The "Conclusions" subsection provides the overall rationale for the Hope Creek "Response to Question B".

II. Process Controls

The "Response to Question A" section of this document provides details of the processes which may affect operating, maintenance, or testing procedures and the processes by which procedures are controlled. The "Response to Question D" section of this document describes the corrective action program and provides an assessment of its effectiveness. The corrective action program establishes the means by which conditions adverse to quality, including those related to procedures, are identified, evaluated and corrected. This section, "Response to Question B," further describes the processes and presents the process effectiveness, as they relate to the translation of design bases requirements into operating, testing, and maintenance procedures.

The operating license for the HCGS is based on the Final Safety Analysis Report which contains design bases information for the facility. As part of the initial licensing of HCGS, as documented in the NRC Safety Evaluation Report related to the operation of HCGS (NUREG-1048) dated October 1984, the NRC stated that PSE&G's program for use of operations and maintenance procedures meets the relevant requirements of 10CFR50.34 and is consistent with the guidance in RG 1.33, Revision 2.

The HCGS Technical Specifications (TS) comprise Appendix A to the respective unit's operating license. The TS contain safety limits and limiting conditions of operation (LCOs). The TS action statements prescribe the measures to be taken in the event the LCOs are not met, in order to maintain the plant in safe condition. The TS prescribe periodic surveillance requirements to demonstrate the operability of SSCs subject to the LCOs.

Implementation of TS limiting conditions of operation (LCOs) and surveillance test requirements is a critical aspect of maintaining plant operations, maintenance and testing activities consistent with the design bases. Maintaining cognizance of plant equipment status and ensuring LCO compliance are primary responsibilities of the Operations Department. The Work Control Process is used to schedule periodic TS surveillances with a nominal scheduled interval of one week or greater. For conditional TS surveillances (e.g., those which are only required when certain equipment is inoperable) and TS surveillances with a nominal interval less than one week, the activities may be tracked to completion using action requests (ARs) within the Managed Maintenance Information

System (MMIS) and/or logs and checklists used by the responsible implementing departments (e.g., Operations, Maintenance). The license amendment process includes the development of implementation plans to identify the procedures affected by proposed amendments and coordination of the necessary changes upon NRC approval and implementation.

The design and configuration change processes can have a impact on station procedures. Procedures and procedure revisions required for the implementation of a design or configuration change are identified during the development and review of the change package. The Modification and Testing section of the change package includes testing requirements necessary to demonstrate the ability of the affected structures, systems and components (SSCs) to satisfy design bases parameters and attributes affected by the change. The close-out section of the change package includes the identification and tracking of procedure revisions requiring implementation prior to returning the SSCs to service, including surveillance testing, corrective maintenance or system operating procedures. Controls placed on the affected procedures via the change package ensure that configuration control is maintained during and after change package implementation.

The UFSAR update process implements 10CFR50.71(e), which requires that the UFSAR accurately reflect the effects of changes to the facility and procedures. The 10CFR50.59 process is an integral part of the UFSAR update process, in that 10CFR50.59 provides the mechanism for evaluating proposed activities against the design bases in the UFSAR. The Nuclear Procedure System establishes requirements for procedure revision and use as described in "Response to Question A". Regardless of the source of a change to a procedure (e.g., design change package (DCP), license amendment, stand-alone procedure change to facilitate a plant activity), the procedure is subject to the procedure control process, which invokes the requirements of 10CFR50.59. The UFSAR update process, 10CFR50.59 implementation and the Nuclear Procedure System are interdependent processes which collectively provide reasonable assurance for the translation of the plant design bases into operating, maintenance and testing procedures.

Inconsistencies between procedures and the design bases may be identified during routine activities (e.g., preparation of a 10CFR50.59 evaluation for a design change), by internal review activities, or by independent assessments. In any case, the corrective action program (described in "Response to Question D") provides a mechanism for the identification, evaluation, trending and correction of conditions adverse to quality, including those related to operating, maintenance and testing procedures. In addition to identifying and correcting

individual discrepancies between the design bases and procedures, the corrective action program enables PSE&G to address potential programmatic deficiencies through the use of trending, root cause analysis and monitoring the effectiveness of corrective actions.

III. Validation Reviews

Over time, PSE&G has conducted a number of broad scope reviews that have assessed whether the design bases were adequately translated into implementing procedures. Included are reviews conducted during initial licensing as well as more recent efforts, several of which are still in progress. A brief description of each effort, the results the relevant conclusions is provided below.

A. Safety System Functional Review (SSFR) / System Functional Review (SFR)

Description of SSFRs/SFRs

Starting in 1987, PSE&G initiated a series of Safety System Functional Reviews (SSFRs) and System Functional Reviews (SFRs). The SSFRs and SFRs were conducted in a manner similar to the NRC's SSFIs. The objectives of an SSFR/SFR is to assess the operational readiness of a selected safety system. The following objectives relate to verifying the design bases was translated into the operations of the facility:

- Testing is adequate to demonstrate that the systems would perform the safety functions required.
- Human factors considerations relating to selected systems (e.g., accessibility and labeling of valves) and the supporting procedures for those systems are adequate to ensure proper system operation under normal and accident conditions.
- Management controls including procedures are adequate to ensure that safety systems will fulfill the safety functions required by their design bases.

The review team first identified the design bases of the system, initiated Technical Review Plans (TRPs), and initiated data gathering activities. The TRPs were prepared to serve as guidance to the review team about the scope and detail of the review and also to highlight potential problem areas identified by PSE&G. TRPs were prepared for each major functional area including,

Maintenance, Operations, Design Change Process, Quality Assurance, Human Factors and Training.

Results of SSFRs/SFRs

The following is a listing of the reviews performed at Hope Creek:

- Station Service Water System (SSWS) SSFR was completed on September 1987.
- Pneumatic Systems SFR was completed on June 2, 1989.
- Station Auxiliary Cooling System (SACS) SSFR was completed in December 1990.
- Primary Containment Isolation System (PCIS) SSFR was completed on November 1, 1993.
- Residual Heat Removal (RHR) SSFR was completed on June 8, 1995.
- Radwaste SFR was completed on August 31, 1995.

These efforts provide a historical perspective and were performed using the industry standards at the time of each review. It is recognized that subsequent to these efforts issues related to some of the systems reviewed have been raised. However the SSFR/SFR review efforts did evaluate aspects of the translation of the design bases information into the operations of the facility and provides a level of assurance in assessing that the process is effective.

A summary of the results of the various efforts is provided below.

SSWS SSFR (1987)

In September of 1987, PSE&G's On Site Safety Review Group and Cygna Energy Services completed a three month detailed inspection of the SSWS. The following objectives and results related to translation of the design bases to the operation of the facility were noted:

- The inspection team examined a number of corrective maintenance, preventive maintenance and surveillance procedures for adequacy and accuracy and completed this review by physically inspecting a sample of components upon which maintenance had been recently applied. In addition, the review found that vendor manual references were applicable, accurate and current. The review team found a corrective or preventive maintenance procedure in place for important SSWS components and the overall format

and content of the procedures to be satisfactory. The review of surveillance test procedures found minor errors but did not yield any deficiencies that were judged to be safety significant.

- Normal, Abnormal, Alarm Response, and Emergency Operating Procedures associated with SSWS were reviewed for technical correctness. The review team found that the procedures reflected the present plant configuration, were technically adequate in all respects and provided the operator with sufficient and clear direction.

In 1987 the overall conclusion of the review was that no critical design, maintenance, programmatic, or operational problems were identified that could be classified as safety significant or required immediate attention. In addition, the operational readiness of the SSWS was confirmed and that the operational programs in place to support SSWS operation were effective and being implemented properly.

PNEUMATIC SYSTEMS SFR (1989)

In May of 1989, PSE&G's On Site Safety Review Group and Cygna Energy Services completed a twelve week detailed inspection of the Pneumatic Systems. The systems reviewed were the Service Air System, Instrument Air System, Primary Containment Instrument Gas and the Emergency Diesel Starting/Control Air System.

- The inspection team sampled a number of corrective maintenance, preventive maintenance and surveillance procedures for adequacy and accuracy. Concerns were identified with frequency of testing for dewpoint and the technical adequacy of that measurement. These issues were evaluated and corrected in accordance with the corrective action program. Generally the surveillance testing was found to be done accurately, however, on case was noted where the procedure did not meet technical specification surveillance acceptance criteria. This issue was corrected by revising the surveillance procedure to adequately reflect the acceptance criteria.
- Normal, Abnormal, Alarm Response, and Emergency Operating Procedures for the associated pneumatic systems were reviewed for technical correctness. The reviewed procedures were found to be technically adequate with a few deficiencies; the team concluded with reasonable assurance that the procedures adequately support proper plant operation.
- A review of Design Change Packages (DCP) was conducted to ensure that design base impacts were properly evaluated , that all necessary reviews and post installation testing was carried out and that all documentation effected

by the modification was updated. The team performed a detailed review of 17 DCPs and no observations or findings were noted in the procedural revision area.

In 1989 the SFR team concluded that the systems reviewed would function adequately and would perform their intended functions under normal and abnormal conditions. The team noted cases of procedure inadequacies with relation to the design bases. Each of these observations were evaluated and resolved by the corrective action program.

<u>SACS SSFR (1990)</u>

A SSFR was conducted from September 17 to October 19, 1990 on the SACS. The team was comprised of members from the PSE&G's On Site Safety Review Group, United Energy Services and Gilbert Commonwealth. The following objectives and results related to translation of the design bases to the operation of the facility were noted:

- In the operations area, an assessment of the adequacy of operational activities relating to SACS was conducted, including a review of operational procedures. All Normal, Abnormal and Annunciator Response Procedures associated with SACS were reviewed for technical correctness by determining if the SAC system was being operated within its design limitations. The review noted the following findings related to the translation of the design bases into the operation of the facility:

  1. The review noted that SACS was operated below the minimum design temperature of the system. Subsequent engineering calculations demonstrated that operating SACS as low as 32°F is acceptable.
  2. The abnormal operating procedure allowed continued SACS operation at temperatures 20°F above the maximum design bases temperature of 95°F. The affected procedure was revised to reflect the 95°F maximum design temperature.
  3. During a LOP with control room abandonment, it appeared to be difficult for operators to automatically control makeup to each SACS expansion tank. This issue was evaluated and resolved by the operations department.

  The team concluded that overall the SACS is being operated under normal conditions according to written approved and generally acceptable procedures. The procedural deficiencies identified during the SSFR were not

considered safety significant and did not have an immediate impact on SACS operation.

- In the testing area, an assessment was conducted to ensure the adequacy of preoperational, surveillance and post-maintenance and modification testing. Overall, surveillance and inservice testing was found to be performed completely, accurately, however the team concluded that surveillance testing procedures should be improved to clearly demonstrate functional operability of the SACS components and the acceptance criteria in these procedures should be reviewed for adequacy in verifying Technical Specification requirements. The procedure deficiencies were forwarded to operations and were subsequently resolved by revising the affected procedures.

In 1990 the teams overall conclusion was that the results of the review, in most cases, confirm a sound design and indicate that the SACS will perform its intended function.

PCIS SSFR (1993)

From June 1 to July 9, 1993, PSE&G and United Energy Services performed a detail inspection of the PCIS and the Nuclear Steam Supply System Shutoff System (NS4). The inspection was organized into five functional areas. The following review areas provide information related to the translation of the systems design bases into the operation of the facility:

- In the Maintenance functional area, maintenance procedures were reviewed to ensure that vendor technical manual recommendations were appropriately addressed. The review team found that HCGS adequately implemented vendor recommendations and that the procedures were considered adequate to perform maintenance activities.
- In the Testing functional area, the adequacy of preoperational, surveillance and post-maintenance testing was assessed. The team reviewed 6 of the 7 preoperational tests related to PCIS and NS4. The team concluded that the scope and detail of the preoperational test meet the UFSAR Chapter 14 and Regulatory Guide 1.68 requirements relating to preoperational testing and that all test objectives and acceptance criteria were met. In the post-modification testing area the team had a concern with adequacy of the stated testing requirements. These concerns were raised to management and received an operability determination and were not judged to affect the operability of the system. Forty seven technical specification surveillance procedures were reviewed. With some exceptions, (i.e., minor deficiencies that did not impact the operability) the team found the test procedures

technical content, test objectives and acceptance criteria adequate to verify system operability.

- In the Operations functional area, the adequacy of normal, annunciator response, off-normal and emergency procedures were reviewed. The review was to determine if the PCIS was being operated within design limitations and whether operational activities were being conducted with adequate procedural guidance. Operations Department emergency, abnormal and normal operating procedures were found to be adequate and provided adequate guidance to operate PCIS during various plant conditions. This was based on an error free walk-through of the reviewed procedures.

In 1993 the team concluded that the PCIS structures, systems and components were designed, installed, tested, operated, maintained, modified and managed such that the system is expected to perform its intended safety function.

RHR SSFR (1995)

During the period of March 6 to April 13, 1995 PSE&G performed a detailed inspection of the Residual Heat Removal System. The inspection was organized into three functional areas. The following review area provided information related to the translation of the systems design bases into the operation of the facility:

- In the Operations functional area, the team assessed the adequacy and accuracy of procedural content. The team reviewed a sampling of normal operating, abnormal, integrated and emergency and surveillance procedures. The team found multiple examples of procedures which failed to provide sufficient, or in some cases, accurate information to the operators. The procedures were revised to provide additional cautions or guidance to the operations staff in order to correct the identified concerns.
- In the maintenance are, the team reviewed selected maintenance procedures to ensure the applicable vendor manual instructions were correctly incorporated into the implementing procedures. The team concluded that the procedures were maintained current with the latest revision of the vendor manuals. The deficiencies noted in this review were determined not to impact RHR system performance and were resolved with the sites corrective action program.

The procedural errors were noted during the SSFR review were evaluated to not impact the operation of the RHR System. These items were corrected in accordance with the corrective action program. The team concluded that, except

for the apparent inability to implement the fuel pool cooling assist mode as designed, the RHR system is capable of meeting its intended function. The inability to implement the RHR mode of fuel pool cooling assist was not significant at that time since other methods are available to provide for fuel cooling and the fuel pool cooling assist mode is only required after the 13th refueling outage, concurrent with a full core offload.

RADWASTE SYSTEMS SFR (1995)

The PSE&G Safety Review Group completed a detailed inspection of the Radwaste Systems in August 1995. This review evaluated the gaseous, liquid, and the solid radwaste systems. This results of this effort was reviewed to determine the effectiveness of translation of the design bases into the operating, testing and maintenance procedures. Findings related to procedures and plant operation included:

- Liquid Radwaste Systems were not being operated consistent with their design bases. This included inconsistencies between procedures and the vendor manual regarding actions for addressing a charcoal bed fire, radwaste tank drain sump that are normally open in accordance with procedures when the design bases requires the valves be closed for room flooding considerations; and operation of the Liquid Radwaste System and the Liquid Effluent Radiation Monitoring System in a manner different from that described in the FSAR.
- The team identified a number of errors, inconsistencies and other weaknesses in operating procedures, alarm response procedures and surveillance procedures.
- Weaknesses in communication between groups led to differences between operating procedures and the system design bases.
- Lack of suitable management oversight led to a number of instances in which equipment or procedures did not match the UFSAR and for which the team did not identify suitable UFSAR change requests.

Overall, the SSFR team found that the Gaseous Waste Management System (GWMS) was capable of performing its functions and satisfying design requirements. However, the Liquid Waste Management System (LWMS) was determined to perform significantly below its design objectives.

Conclusions

The SSFR/SFR efforts represent a number of comprehensive reviews. The effort included elements that assessed the accuracy of the implementing controls and procedures as they relate to the design bases. The NRC has reviewed the findings of the RHR SSFR and Radwaste Systems SFR and have found the efforts to be comprehensive.

With the exception of the Radwaste Systems SFR, the SSFRs/SFRs performed since 1987 did not identify any weakness that would have challenged system operation, or the ability of the system to perform its intended function. Generally the design bases were adequately found to be translated to operation, maintenance and testing procedures.

B. Technical Specification Surveillance Improvement Program (TSSIP)

Description of TSSIP

In response to Hope Creek Surveillance program weaknesses noted in the early 1995 time frame, Hope Creek station management instituted the Technical Specification Surveillance Improvement Program (TSSIP). The purpose of the TSSIP effort was provide reasonable assurance that the surveillance program at Hope Creek was adequate. The effort began in October 1995 and the reviews completed on December 31, 1996.

TSSIP did a thorough review of the surveillance program including verifying the scheduling mechanisms and ensuring that the actual implementing procedures to verify numerical values, setpoints, tolerances and calculations referenced in the implementing controls met the acceptance criteria stated in the technical specification surveillance requirement.

Results of TSSIP Program

Performance of the TSSIP project was done in two phases. Phase I was a global review to ensure that all surveillance items had an associated implementing control and that the scheduling mechanism was accurate. Phase II was a detailed review of the specific implementing controls (i.e., surveillance procedures) to verify their consistency with the design bases, as represented in the HCGS Technical Specifications.

The primary focus of the Phase I review was the Technical Specification Surveillance Matrix, maintained by the technical specification coordinator and the Recurring Task numbers used by the Managed Maintenance Information System (MMIS) used to generate Work Orders for the performance of the actual work. The Technical Specification Cross Reference Matrix (TSCRM) is a document that cross references the surveillance requirement against the surveillance procedures and the recurring tasks associated with each procedure. A number of deficiencies in the TSCRM and Recurring Tasks were noted during this review and were forwarded to the station for resolution. However none of the deficiencies challenged the operability of the equipment. The Phase I review was completed and provided assurance that the surveillance program had an implementing control for each surveillance requirement and that the scheduling tools were adequate.

A detailed review of the implementing surveillance procedures was conducted during the detailed Phase II review. This review was conducted for each individual surveillance requirement. There were 782 surveillance requirements and these were organized into 170 review packages. The packages provide a hard copy result of the reviews and contain the applicable technical specification sections, the implementing procedures, the results of the reviews, including any action requests, procedure revision requests, or changes to the MMIS recurring tasks, as well as marked up drawings and other supporting information. During the review 70 findings related the technical specification surveillance program were identified and entered into the NBU corrective action program. Each of these findings were evaluated for safety significance and reportability. Of these 70 issues, 15 were considered reportable as missed or inadequate surveillance procedures that did not adhere to the technical specification acceptance criteria. Most of these findings were reported in LER 354/95-33 and its associated supplements. Only one of these findings, dealing with the RHR heat exchanger bypass valve, actually failed the revised surveillance test when the surveillance test was run. This was due to a partially dislodged relay. The relay was subsequently cleaned and replaced and the test was satisfactorily completed. The issues were identified and reported in accordance with the Corrective Action Program to the operations staff.

The TSSIP process was periodically reviewed by the NRC particularly during the Readiness Assessment Team Inspection at the conclusion of RF06. In addition the NRC noted this effort as a positive effort in the latest Systematic Assessment of Licensee Performance (SALP) report issued December 24, 1996.

Conclusions

The TSSIP review effort has been completed at Hope Creek. The issues identified during the TSSIP review were entered in the NBU corrective action program and resolved or resolution is being tracked in accordance with the corrective action process.

This effort constituted a detailed review of the Hope Creek surveillance program and its associated surveillance procedures. This program has verified that the surveillance procedures adequately implement the TS surveillance requirements. These requirements are the parameters necessary to ensure operability of the facilities SSCs in accordance with its design bases. Therefore this effort has effectively performed a validation of the Hope Creek surveillance program and has provided a high level of assurance that the program is adequate in its implementation.

C. Review of Hope Creek Procedures

Description of Review

In December 1996, a sample of Hope Creek procedures were reviewed to assess consistency with the current plant licensing and design bases. The population of reviewed procedures included a sample of Operations, Maintenance, Radiation Protection, Chemistry and Fire Protection Department procedures. Each department reviewed their own procedures with the exception that the System Engineering Department assisted in the review of the Operations Department procedures. The procedure reviews ensured that the design bases information, as reflected in the Updated Final Safety Analysis Report, Technical Specifications, and NRC Safety Evaluation Reports were adequately and accurately portrayed in the reviewed procedures. The Hope Creek On-Site Safety Review Group (SRG) performed an independent assessment of the Operations and Maintenance Departments review effort.

Results of Review

Fourteen Operations Department procedures were reviewed. The review identified that eleven of the fourteen procedures required revision to address minor discrepancies and inconsistencies with design and licensing basis documents. None of the inconsistencies affected the operability or functionality of the system. The inconsistencies are being addressed in accordance with the PSE&G Corrective Action Program. The procedure review did point out a programmatic weakness in the operations administrative procedure relating to

operability determinations. HC.OP-AP.ZZ-108, "Operability Assessment and Equipment Control Program", needs to be modified to specify when a 10CFR50.59 review is required for compensatory measures that are being implemented to support operability of a system, structure or component in a degraded condition. The revision to HC.OP-AP.ZZ-108 is currently in progress and is being tracked by the corrective action program.

The Maintenance Department reviewed sixteen procedures, including the two Maintenance Department troubleshooting procedures. As a result of this review, only minor typographical and editorial errors were identified in the non-troubleshooting procedures. With respect to the troubleshooting procedures, a programmatic issue relating activities requiring 10 CFR 50.59 reviews was noted. Specifically, the troubleshooting procedures rely solely on the shift technical advisor (STA) review to determine if a test or experiment is involved. The STA position may be filled by an individual not trained to perform 10CFR50.59 reviews and no peer review is performed as is the case for other 10CFR50.59 applicability reviews. As a result of this finding, the troubleshooting procedures will be revised to ensure the design and licensing bases are preserved throughout the troubleshooting process. For troubleshooting activities that maintain and/or restore operability of safety related SSC's, a 10CFR50.59 review will be performed. These procedure revisions are being tracked by the corrective action program.

The Radiation Protection Department conducted a review of three procedures. Only discrepancies, that did not impact operability, between the procedures and the licensing and design bases information were identified as a result of this review.

The Chemistry Department conducted a review of eight procedures. Minor discrepancies between four procedures and the licensing and design bases information were identified as a result of this review and are being resolved using the PSE&G Corrective Action Program. Chemistry plans to conduct a review of the UFSAR to determine if chemistry-related requirements are appropriately incorporated into plant procedures.

The Fire Protection Department conducted a review of 8 procedures. Minor discrepancies between the procedures and the licensing and design bases information were identified as a result of this review and are being resolved using the PSE&G Corrective Action Program.

The SRG performed an independent assessment of the reviews conducted by the Operations and Maintenance Departments. The SRG operations review

verified the conclusions for two of the 14 procedures sampled. The SRG confirmed the results of the operations review and stated that the procedures reviewed accurately reflected relevant Technical Specification and UFSAR requirements. The SRG review of 3 Maintenance procedures also noted the procedures reviewed accurately reflected any relevant Technical Specification and UFSAR requirements. The SRG review also noted the concern related to troubleshooting procedures described above.

Conclusions

The review found the procedures reviewed to be adequate to support operations and no procedural issues were identified that challenged the operation of the facility in accordance with its design bases. The review did reveal inconsistencies between the UFSAR and the procedures reviewed, but the identified discrepancies were evaluated to be minor in nature. However a process weakness related to procedure reviews in accordance with 10CFR50.59 was noted. This is discussed in further detail later in this response. Future activities in this area are outlined in Attachment 2.

D. Configuration Baseline Document (CBD) Validation Program

Description of the CBD Validation Project

The CBD validation project is an ongoing effort that ensures that design output and operating documents (e.g., UFSAR, TSs, P&IDs, operating and emergency procedures, test procedures, I&C program and ASME Section XI program) are consistent with the design bases. The scope of the project currently includes six Hope Creek systems. The Station Service Water System and the Station Auxiliary Cooling System reviews has been completed and the High Pressure Coolant Injection System and Emergency Diesel Generator System reviews are currently in progress.

Station Service Water System Collegial Self-Assessment (SA-96-012)

This CBD validation was performed on the SSWS and the Service Water Traveling Screens in April and May 1996. The SSWS Chlorination system was not reviewed.

During the SSWS review, weaknesses were identified relative to: 1) lack of consideration for instrument inaccuracies in design analyses, surveillance testing and TS limits and 2) inconsistencies between procedures and design calculations; 3) inconsistencies between Alarm Response procedure setpoints. However, these issues were evaluated and determined not to challenge the

operability of the SSWS. These issues have been or are being resolved and tracked in accordance with the corrective action program.

Safety & Turbine Auxiliaries Cooling System (STACS) Collegial Self-Assessment (SA-96-015)

In June through August 1996, CBD validation was performed on the STACS. The non-safety related Turbine Auxiliaries Cooling (TACS) portion of the STACS system was not reviewed.

The following weaknesses were identified: 1) original plant design calculations did not accurately reflect the as-built configuration, configurations allowed by operating procedures, or the appropriate effects of a loss of instrument air following a Loss of Coolant Accident/Loss of Offsite Power; and 2) maintenance of design bases documentation, such as design calculations and CBDs, appeared to be weak. Findings included concerns relative to the bases for the extended allowable out-of-service times for the SSWS, SACS and EDGs, and identification of inadequacies in the SACS startup testing. Deficiencies were logged and tracked by the corrective action process, closure of these items is in progress

Conclusion

The completed CBD reviews of the SSWS and the SACS have identified some procedure inconsistencies with design bases documents, however the safety impact has been evaluated in accordance with the corrective action program, and found not to impact system operability. The weaknesses that have been identified, are being resolved and tracked through the corrective action process. The CBD validation process is still ongoing as described in PSE&G's future actions outlined in Attachment 2.

E. Accident Analysis (UFSAR Chapter 15) Review

Description of Review

An effort was conducted to identify and validate a sampling of salient parameters and assumptions from the Hope Creek accident analysis as presented in Chapters 15 (Accident Analysis) and Sections 6.2 (Containment Response) and 6.3 (DBA LOCA Response). The purpose is to ensure that the parameters stated in the UFSAR Accident Analysis remain consistent with the original design bases for the bounding accidents and that the stated limits and parameters are adequately maintained and implemented via the stations operational controls. These parameters will be compared against the source inputs available from the

vendor and the facility implementation documentation, consisting of the facilities Technical Specifications, operational procedures, controlled drawings and calculations.

In addition to this effort, by the nature of the nuclear physics characteristics of a BWR core, a cycle specific analyses is performed in which non-bounding safety analysis are reanalyzed. PSE&G engages in a formal reload process with the fuel vendor in accordance with written guidelines provided by the fuel vendor. The vendor guidelines provide a formal communication vehicle on the plant characteristic parameters that are vital in the transient analyses for reload licensing applications and assures that the design and or configuration changes are included in the applicable analyses. This data transfer to the fuel vendor is divided into two sections, with the first section including key parameters that are directly applicable to the reload analysis and the second section including those parameters that are applicable to the verification of the remaining UFSAR events. Due to the cycle specific nature of the reload analysis, this process evaluates and updates if necessary the limiting transients analyzed in the UFSAR. This process is in essence a reassessment of the transient analysis that is conducted on a per-cycle basis.

Results of Review

Identification of the accident analysis parameters has been completed, identifying just over 400 review items. The input assumptions are listed for both design and performance parameters. The design parameters are those inputs used in the accident analysis and the performance based parameters represent the current facility performance characteristics used in the cycle specific analysis to provide better modeling.

Of the initial parameters, 213 were considered to be directly translated via the facilities technical specification limits (i.e. safety related instrument and protection system setpoints, ECCS flowrates, etc.). Based on the current Technical Specification Surveillance Improvement Project, recently completed at Hope Creek, PSE&G has assurance that these values are adequately translated and implemented. One hundred thirty-three of the values were determined to be design inputs that require verification by review of the physical plant configuration. The review of these values will be conducted as part of the design bases review project discussed in Attachment 2.

Conclusion

The major portion of this review has been completed with a large number of the items identified being directly tied to values dictated by the facilities Technical

Specifications. Based on the controls needed to change the Technical Specifications we have a high level of assurance in the values stated in technical specifications. The TSSIP effort has recently validated the adequacy of the technical specification surveillance program and has corrected all identified deficiencies, therefore there is a level of assurance in the values associated with the facilities technical specifications. The review of the remaining items is on-going and will be rolled into our design bases review project discussed in Attachment 2.

In addition to this review of the implementing controls, a review of the fuel vendor analysis is performed each cycle to ensure that the values are still consistent with the initial licensing analysis. In this way, PSE&G has added assurance that the accident analysis parameters covered by these implementing controls are indeed the values used by the fuel vendor in the cycle specific analysis.

IV. Assessment Of Effectiveness

This section provides an assessment of the effectiveness of key processes used to ensure that: 1) the design bases are translated into operating, maintenance and testing procedures, 2) the plant configuration is maintained consistent with the design bases during the use of operating, maintenance and testing procedures and 3) testing procedures are implemented such that the plant configuration and equipment performance are maintained consistent with the design bases. As part of the discussion of the nuclear procedure system, an assessment is also made relative to the current adequacy of operating, maintenance and testing procedures. This section uses the organizational framework of the "Response to Question A".

A. Design and Configuration Change Processes

The design change process affects translation of the design bases into operations, maintenance and testing procedures in that the process must identify procedure changes required as a result of a design change. The DCP process also contains requirements to ensure that proper post-modification testing is completed such that the plant configuration and equipment performance are maintained consistent with the design bases.

<u>*Post-Modification Testing*</u>

A review of recent internal and external assessments was conducted to measure the effectiveness of post-modification testing in determining that the design requirements were satisfied. Relevant information was noted in NRC inspection

reports and a third party assessment of DCP quality conducted in October and November 1995.

Based upon the assessments as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. An inadequate post modification test in 1992 resulted in mis-positioning the SSWS/SACS throttle valves.
2. Inadequate post modification testing of the digital feedwater system modification contributed to two reactor scrams in 1994.
3. Post-modification testing concerns were identified relative to two of the seventeen DCPs reviewed during the October/November 1995 third party assessment. Two issues were identified associated with the first DCP and three issues were identified relative to the second DCP. One of the issues associated with the first DCP was dispositioned by PSE&G as requiring no change to the testing while the recommendation associated with the second issue was accepted. The recommendation for enhanced testing had no impact on the operability or functionality of the system. The second DCP was never issued and the identified post modification test issues were therefore not dispositioned and resolved.

Based upon the above information, PSE&G concludes that reasonable assurance exists that proper post-modification testing is completed such that the plant configuration and equipment performance are maintained consistent with the design bases.

Procedure Changes Resulting From DCPs

A review of recent internal and external assessments was conducted to determine the effectiveness of the DCP process in identifying necessary procedure changes. Relevant information was noted in NRC inspection reports and the following independent assessments: 1) a third party assessment of DCP quality conducted in October and November 1995 and 2) a third party assessment of DCP quality conducted between November 1995 and April 1996.

Based upon the assessments as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. The DCP for the hardened plant vent modification was complete and contained appropriate recommendations for changes to plant procedures.

2. The reactor vessel water level instrumentation backfill modification was adequately designed, installed and tested and the operating procedures were in place and contain adequate directions to support system operation.
3. A third party review identified one enhancement to a procedure change implemented as a result of a DCP.
4. Hope Creek was found to have an effective process for identifying DCP related procedure changes.

Based upon the above information, PSE&G concludes that there is reasonable assurance that required procedure changes will be identified by the DCP process.

B. Configuration Control Process

The nuclear procedure system, work control, safety tagging, post-maintenance testing/operability re-testing (PMT/RT) and the TS surveillance program are the processes described in this section. The nuclear procedure process directly affects translation of design bases requirements into procedures in that it governs the development of new procedures and revisions to existing procedures. While the other processes discussed in this section do not directly affect translation of the design bases into procedures, they are relevant because they establish controls on the implementation of work activities governed by station procedures. The work control and safety tagging programs serve to maintain the plant configuration consistent with the design bases during the use of operations, maintenance and test procedures. The PMT/RT program and the TS surveillance program serve to ensure proper implementation of test procedures such that the plant configuration and equipment performance are maintained consistent with the design bases.

Nuclear Procedure System

This section includes assessment information for the nuclear procedure system and its implementation as well as the adequacy of the existing procedures.

The program that governed the initial development of Hope Creek station procedures used Regulatory Guide 1.33, Revision 2 and ANSI N18.7-1976/ANS 3.2 (substituting Section 5.3 of ANSI/ANS 3.2-1982 for Section 5.3 of ANSI N18.7-1976/ANS 3.2). As documented in NUREG-1048, the NRC reviewed and accepted the program for procedures other than EOPs. Documentation of the

review of the development process for the EOPs is contained in Supplement 5 to NUREG-1048.

A review of recent internal and external assessments was performed to evaluate the effectiveness of the procedure development and revision process, its implementation, as well as the adequacy of procedures. Relevant information was noted in NRC inspection reports and the following internal assessments: 1) the March/April 1995 RHR SSFR which is described in the "Validation Reviews" subsection, 2) the August 1995 radwaste SSFR which is described in the "Validation Reviews" subsection, 3) an Onsite Safety Review Group common cause assessment of EDG events and failures completed in March 1996, 4) observations by senior management in March 1996, 5) an assessment by a procedure upgrade team in response to the March 1996 senior management observations 6) the December 1996 review of a sample of Hope Creek procedures that is described in the "Validation Reviews" subsection and 7) a Licensing and Regulation Department common cause assessment of 10CFR50.59 reviews of Hope Creek procedure revisions that was completed in January 1997. Additional procedure related issues were identified through the corrective action program.

Based upon the above assessments, the following general findings were identified:

1. Some weaknesses in surveillance test procedures were identified in early 1995 with additional examples identified leading up to and during implementation of TSSIP. Several of these resulted in NRC violations.
2. Procedure problems contributed to many of the EDG events and failures that occurred between January 1, 1995 and February 5, 1996.
3. Most problems with procedure adequacy have been minor in nature; however, procedure deficiencies have either caused or contributed to a number of Hope Creek events and issues. These include the April 1995 radiation release, the July 1995 shutdown cooling bypass event, the improper procedural limits on rod speed testing, the August 1996 SACS deficiencies and the October 1996 issue involving the valve from the standby SACS loop to the RHR heat exchanger being open for minflow purposes.
4. Discrepancies were identified between some procedures and setpoint data (ICD cards) that were attributed to inadequate implementation of the procedure change process.

5. Process enhancements were identified relative to NAP-1 and the writers guide.
6. Concerns related to the implementation of the 10CFR50.59 process for procedure revisions were identified and are addressed in detail later.

Procedure inadequacies have been created during development and revision of procedures. Weaknesses in implementation of the 10CFR50.59 process that will be discussed later have contributed to the inadequacies. Some of the above examples associated with the adequacy of procedures have resulted in challenges to equipment operability. Actions have been taken or are planned to improve in the nuclear procedure system and its associated procedures. These include the following:

- A review is being completed to determine if a programmatic concern exists relative to agreement between procedures and setpoint data (ICD cards).
- A procedure upgrade team was assembled to evaluate the governing procedures used to control the development, revision, review and content of procedures.
- In December 1996, revisions to NAP-1 and NC.NA-WG.ZZ-0001(Q), "Procedure Writers Guide" (NWG-1) were issued. Key elements of the procedure control process per NAP-1 and the writers guide that affect translation of design bases information into procedures included the following: 1) requiring new and revised implementing procedures and "Q" designated administrative procedures (i.e., those procedures for safety related activities in Regulatory Guide 1.33) to receive a 10CFR50.59 applicability review, 2) requiring revisions to "Q" designated implementing procedures and implementing procedures described in TS 6.8.2 to be reviewed by a Station Qualified Reviewer (SQR), 3) incorporating SQR qualification requirements into NAP-1 and 4) adding new sections to the NWG-1 that discuss procedure bases, general requirements and the procedure Verification and Validation.
- TSSIP has been completed to provide reasonable assurance of the adequacy of surveillance test procedures.
- Additional activities are planned to provide increased assurance in operating, maintenance and other testing procedures. These activities are described in Attachment 2.

Some weaknesses have been identified in the process for creating and maintaining procedures. NAP-1 and the writers guide have been revised to

address these process concerns. Since these revisions were only recently implemented, the effect of the changes on the content of procedures in the NBU has not been measured. PSE&G believes that the process has been enhanced such that the process concerns that led to the problems have been addressed. The 10CFR50.59 implementation concerns will be discussed in greater detail later.

The TSSIP has provided provide reasonable assurance that design bases as stated in the TS has been translated into surveillance test procedures. With respect to other operating, maintenance and testing procedures, PSE&G has reasonable assurance that the design bases has been translated such that continued plant operation is supported. PSE&G also has reasonable assurance that the corrective action program coupled with the additional procedure review activities discussed in Attachment 2 will identify any existing discrepancies between procedures and the design bases and that appropriate actions will be taken in accordance with the corrective action program and plant license if such discrepancies are discovered.

Work Control

A review of recent internal and external assessments was conducted to determine the effectiveness of the work control program and its implementation. Relevant information was noted in NRC inspection reports and the following internal assessments: 1) the QA startup progress assessment completed in February 1996 and 2) numerous self assessments performed by the Maintenance Department and Planning Department.

Based upon the internal and external assessments as well as evaluation of issues identified through the Corrective Action Program, general findings were as follows:

1. In late 1995, weaknesses were noted in the scheduling of activities in that several were planned coincident with work or degraded conditions on redundant safety systems. In late 1995 and early 1996, several unanticipated events occurred that were the direct result of less than adequate work control administration.
2. The new work week management process implemented in March 1996 has certain logical advantages over the previous scheduling method, especially for controlling the risk of on-line maintenance and is a positive effort to improve the scheduling of activities and to ensure redundant equipment remained available during significant on-line system maintenance outages.

3. There has been mixed performance associated with the work week management process since its implementation in March 1996.

Actions taken or planned as a result of the above include the following:

- In March 1996, Hope Creek implemented a work week management process to schedule and perform work.
- The process will continue to receive frequent self assessments.

The process was considered a weakness by station management and was changed to a work week plan based on successful industry programs in March 1996. Although the performance of the process will continue to be assessed and monitored, based on the above information, PSE&G believes that the existing process provides reasonable assurance that work will be controlled in a manner that will ensure that the plant configuration is maintained consistent with the design bases.

Safety Tagging Program

A review of recent internal and external assessments was performed to determine the effectiveness of the safety tagging program as well as its implementation. Relevant information was noted in NRC inspection reports and in the Hope Creek common cause assessment of tagging errors that was initiated as a result of a May 8, 1996 tagging event. Various safety tagging issues have also been identified through the corrective action program.

These internal and external assessments resulted in the following:

1. The common cause assessment noted personnel performance issues and programmatic deficiencies. Most events analyzed indicated that the program was a complicated process.
2. A non-cited violation (NCV) was identified regarding a failure to implement the station safety tagging program that required a written reply to the NRC. The NRC noted that a written reply to the NCV was required due to the long-standing repetitive nature of tagging problems.
3. The root cause analysis for the safety tagging violations in early 1996 was comprehensive. The frequency of tagging errors has decreased in recent months, indicating that some improvement has already occurred even though all corrective actions have not yet been implemented.

Actions that have been taken or that are planned to improve the safety tagging program include the following:

- Management expectations for implementing the program have been issued
- A desktop guide that establishes clear direction regarding the required elements for tagging preparation and review has been developed
- Guidelines for integrating tagout preparation into the work week process have been developed
- A plan for assessing performance of the program has been developed
- The safety tagging program procedure has been revised to simplify and improve the tagging process by clearly describing the process for adding and removing tags
- An assessment of effectiveness of the new process will be completed by March 1, 1997.

PSE&G implemented process and performance improvement actions to improve the safety tagging program. Improvement has been observed. Although the performance of the process will continue to be assessed and monitored, PSE&G believes that the existing process provides reasonable assurance that work will be controlled in a manner that will ensure that the plant configuration is maintained consistent with the design bases.

Technical Specification Surveillance Program

The internal review effort relevant to the TS surveillance program is the TSSIP program that is discussed in the "Validation Reviews" subsection above. As indicated in that discussion, the effort constituted a detailed review of the Hope Creek surveillance program and its associated surveillance procedures. The review and the resulting corrective actions provided a level of assurance that the program is adequate in its implementation.

In addition, recent improvements have been made to NAP-12 based on lessons learned from TSSIP and prior TS-related events. The revision that includes the improvements is scheduled to become effective on February 14, 1997.

The TSSIP performed a detailed review of the process and that enhancements to the process that incorporate lessons learned from experience have been implemented. These actions provide reasonable assurance that design bases information has been translated into surveillance test procedures and that the TS

surveillance program will ensure proper implementation of the surveillance testing procedures thereby supporting plant operation consistent with its design bases.

Post-Maintenance Testing and Operability Retesting

Based upon review of recent internal and external assessments, relevant information was noted in NRC inspection reports.

Based upon the assessments as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. In March 1996, a near miss occurred relative to failure to adequately conduct post-maintenance scram time testing as required by TSs. This test is a conditional surveillance requirement that must be completed prior to entering Operational Condition 2. Past occurrences were identified in which the testing was not properly conducted in that it was not performed until after entry into Operational Condition 2.

With respect to significance of the above identified issue, PSE&G notes that, for the instances in which the testing was not performed until after entry into Operational Condition 2, subsequent testing demonstrated the operability of the subject control rods.

Actions taken or planned in the area of post maintenance testing and operability retesting are as follows:

- A list of conditional TS requirements has been developed to ensure that sufficient controls are in place to maintain compliance with these requirements during plant operation and outages.
- NAP-50 was revised in December 1996. The technical guidance in the procedure was updated, including the identification of trigger values for specific conditional TS requirements.

Process improvements have been implemented and continued monitoring of performance will be accomplished through the corrective action program. Based on these actions, PSE&G concludes that there is reasonable assurance that the post maintenance testing and operability retesting programs will ensure proper implementation of the appropriate testing procedures thereby supporting plant operation consistent with its design bases.

C. 10CFR50.59 Process

The 10CFR50.59 process directly affects translation of design bases requirements into procedures in that it provides the mechanism for evaluating the impact of changes to plant procedures against the current licensing and design bases.

A review of recent internal and external assessments was performed to evaluate the following elements of the 10CFR50.59 program: 1) procedures and controls, 2) implementation and 3) training and qualification. Relevant information was noted in NRC inspection reports and the following internal assessments 1) an Offsite Safety Review (OSR) Group assessment of 10CFR50.59 applicability reviews completed in November 1995, 2) an OSR Group assessment of 10CFR50.59 safety evaluations completed in November 1995, 3) a Licensing and Regulation (L&R) Department self assessment of the 10CFR50.59 program completed in February 1996, 4) an assessment by an individual from another utility completed in April 1996, 5) an OSR Group assessment of the adequacy of the 10CFR50.59 safety evaluation program completed in May 1996 and 6) an L&R Department common cause assessment of 10CFR50.59 reviews of Hope Creek procedure revisions that was completed in January 1997.

Based upon the assessments as well as issues identified through the corrective action program, the following general findings were identified:

1. Concerns with program implementation were identified in the majority of the assessments.
2. The need to establish qualification requirements for preparers, peer reviewers and approvers of applicability reviews and safety evaluations was also identified.
3. NAP-59 is well written, provides clear assignment of responsibility and provides the user with good directions.
4. The training lesson plans were observed to be clearly written with good examples to demonstrate proper use of the NAP.
5. The 10CFR50.59 program is adequate to support the operation of Hope Creek.
6. An NRC violation was identified in April 1996 relative to use of the 10CFR50.59 process to implement an administrative control on a TS limit. In October 1996, another NRC violation was identified that indicated poor implementation of the program. Another NRC violation was cited relative to failure of OSR to review some safety evaluations.

7. Implementation of 10CFR50.59 with respect to procedure changes is weak in comparison to the plant modification process; it may be too easy to conclude that 10CFR50.59 does not apply, thereby preventing SORC from reviewing the proposed procedure change.
8. A major cause of the inadequate 10CFR50.59 reviews for procedure revisions is personnel performance deficiencies.

The concern with implementation of the 10CFR50.59 program with respect to procedure changes is a common thread running through the recent assessments. The process and implementation concerns associated with the nuclear procedure system governing procedures that were discussed previously coupled with the 10CFR50.59 implementation concerns have resulted in some cases of procedure inadequacy. Some of the inadequacies have resulted in challenges to equipment operability. As a result, actions have been taken or are planned relative to improving the 10CFR50.59 program. These actions are as follows:

- A revision to NAP-59 was issued that added specific training and qualifications requirements for preparers, peer reviewers and approvers of applicability reviews and safety evaluations. The procedures were also modified to be consistent with the NRC interim guidance.
- A 10CFR50.59 newsletter has been issued to 1) increase awareness of 10CFR50.59 problems encountered at Hope Creek and other plants; 2) explain recent revisions made to procedures implementing 10CFR50.59 requirements; and 3) improve understanding of the 10CFR50.59 process.
- Each Hope Creek department is being required to take actions to improve personnel performance in the area of 10CFR50.59 applicability reviews for procedure revisions. Actions to improve personnel performance will be completed by March 1, 1997.

A good framework has been established for an effective 10CFR50.59 program. Recent internal and external assessments have found the program implementing procedures to be well written with good guidance provided to the user. However, recent assessment activities have identified weaknesses in program implementation that are being addressed through the corrective action program. Training is considered adequate and recent improvements in the qualification program will ensure that personnel performing 10CFR50.59 activities have an improved understanding of the regulation. The solid framework currently in place, the ongoing efforts to correct weaknesses in program implementation and periodic assessments of the program provide reasonable assurance that the

10CFR50.59 program is effective in supporting translation of the design bases into operating, maintenance and testing procedures such that continued plant operation can be supported. Effectiveness of the corrective actions will be monitored.

D. License Amendment Process

The license amendment process affects translation of the design bases into operations, maintenance and testing procedures in that the process must identify and ensure implementation of procedure changes required as a result of a license change.

Relevant information was noted in the Licensing and Regulation (L&R) Department self assessment of the TS maintenance program that was completed in February 1996. TS maintenance program issues have also been identified through the corrective action program. The L&R Department self assessment evaluated the effectiveness of the TS maintenance program to support the safe operation of Hope Creek.

Based upon the L&R assessment as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. In late 1995, deficiencies were identified relative to failure of the process to ensure proper and timely identification and processing of UFSAR changes and DCPs needed to support implementation of a number of amendments.
2. In early 1996, an implementation deficiency with another amendment resulted in identification of a generic concern relative to the amendment implementation process at Hope Creek. The specific deficiency involved implementation of an administrative program that compensated for removal of requirements from the TS.
3. The engineering evaluation that supported the license amendment application for License Amendment 75 credited operator actions that were only marginally established in operating procedures.
4. The TS maintenance process is adequate and can support the continued safe operation of Hope Creek.

There have been some problems related to implementation of Hope Creek license amendment; however, only one involved a procedural weakness. Although enhancements to operating procedures were implemented as a result of the weakness, PSE&G notes that the procedures were considered by the Operations Department to be adequate as written. The issue in early 1996

involved an administrative program that could also be considered procedure related.

Actions that have been taken or planned include the following:

- Additional process controls have been incorporated into the program that include a pre-submittal implementation meeting to identify the actions necessary to implement the expected amendment, a post-approval meeting to review the amendment and to finalize the implementation plan and incorporation of revised documents into an implementation package for the concurrence of the Manager - Licensing and Regulation.

Changes have been made to the process to ensure that actions (e.g., procedure changes) that are required to support a license change are appropriately implemented. These actions provide reasonable assurance that changes associated with license amendments will be identified for translation into plant operating, maintenance and testing procedures. PSE&G also has reasonable assurance that changes associated with past amendments have been appropriately translated into operating, maintenance and test procedures such that continued plant operation can be supported.

E. UFSAR Maintenance Process

The UFSAR maintenance process affects translation of the design bases into operations, maintenance and testing procedures in that the UFSAR must be properly maintained to ensure that an accurate document is available when developing new procedures or when evaluating changes to existing procedures.

A review of recent internal and external assessments was performed to determine the effectiveness of the UFSAR maintenance program to support plant operation. The assessments encompassed the program itself as well as its implementation. Relevant information was noted in NRC inspection reports and the following internal assessments: 1) a Licensing and Regulation (L&R) Department self assessment of the UFSAR maintenance program completed in February 1996 and 2) an Offsite Safety Review (OSR) Group assessment of the UFSAR maintenance program completed in May 1996. UFSAR maintenance issues have also been identified through the corrective action program.

Based upon the internal and external assessments as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. A backlog of UFSAR change notices was identified by PSE&G and subsequently cited as a violation by the NRC as a result of Hope Creek UFSAR Revision 6 not reflecting all changes up to a maximum of six months prior to the date of filing.
2. 10CFR50.59 evaluations that identify changes to the plant or to procedures that are described in the SAR are not always accompanied by the necessary UFSAR change notice
3. A need was identified for improving procedural interfaces to ensure required UFSAR changes are properly captured
4. Corrective action program action requests and open items were written against the UFSAR maintenance program; however, most of the issues were administrative in nature and did not impact the technical integrity of the UFSAR.
5. The UFSAR maintenance process was considered adequate to support the continued operation of Hope Creek.

The above assessments have characterized most identified UFSAR problems as not impacting the technical integrity of the UFSAR.

Actions taken or planned as a result of the above include the following:

- The backlog was eliminated. As an interim measure, a list of the backlog and outstanding UFSAR changes had been provided to UFSAR copy holders for use in performing 10CFR50.59 safety evaluations.
- Procedures governing the UFSAR maintenance process have been enhanced. Enhancements include the following: 1) added requirement for the sponsor to coordinate the reviews and obtain required approvals, including SORC approval, 2) made managers responsible to the maintain the accuracy of information in the UFSAR in their area of responsibility; 3) added criteria for monitoring implementation dates for UFSAR changes.
- The procedure governing the 10CFR50.59 process was revised as follows: 1) added a question regarding the need for a UFSAR changes along with a place to document the UFSAR change number on the safety evaluation form and 2) added a requirement to process any related UFSAR change with the primary document being reviewed (e.g., procedure, DCP, etc.).
- A review of TS amendments and related SERs was completed to ensure proper incorporation into the UFSARs.

- Planned improvements to the UFSAR maintenance process include the following: 1) revision to the UFSAR maintenance process description contained in the L&R administrative procedure to make it a sequential process description, 2) incorporation of lessons learned from the June and September 1996 UFSAR revisions into the UFSAR maintenance procedure and 3) completion of a self assessment of the UFSAR maintenance process in 1997.

The UFSAR maintenance process has recently been improved. Procedures have been improved to better integrate plant processes with the UFSAR process. Deviations discovered between the UFSAR and plant configuration are promptly identified and corrected in accordance with the Corrective Action Program. Based on the activities taken to improve the UFSAR process, PSE&G believes that the current UFSAR maintenance process adequately tracks, processes and incorporates identified UFSAR change notices as required by 10CFR50.71(e).

Based on the above as well as the fact that most UFSAR issues identified in the assessments did not impact the technical integrity of the UFSAR, PSE&G believes there is reasonable assurance that the UFSAR is sufficiently accurate to support translation of design bases information into the operating, maintenance and testing procedures.

F. Specific Technical Programs

Inservice Testing Program

As with the PMT/RT program and the TS surveillance program, the IST program serves to ensure proper implementation of test procedures such that the plant configuration and equipment performance are maintained consistent with the design bases.

Based upon review of recent internal and external assessments, relevant information was noted in NRC inspection reports and a QA audit of the program conducted in October 1995. IST program issues have also been identified through the corrective action program.

Based upon the L&R assessment as well as evaluation of issues identified through the corrective action program, general findings were as follows:

1. A procedural inadequacy was identified in that the IST procedures did not require testing of the "A" and "B" SLC pump discharge check valves in the

reverse flow direction. This procedural inadequacy is believed to be an isolated condition based on a review conducted to identify other similar conditions.

2. The QA audit found that most aspects of the program were in compliance with ASME Section XI and NRC Generic Letter 89-04 with the following exceptions:
   - Instances of test methods not meeting regulatory commitments and code requirements
   - Failure to update and maintain some programmatic documentation
   - Lack of administrative control for the purposes of cold shutdown testing
   - Lack of ownership for relief valve testing
   - Ineffective corrective action due to failure to look at widespread generic implications

   The audit determined that increased management attention and oversight was required due to these exceptions.
3. NRC inspections of two implemented modifications (backfill modification and the hardened plant vent modification) have verified that the valves for the installed systems were properly included in the IST program.
4. The latest NRC SALP report indicated that strong support has been provided for programmatic testing activities such as IST.

The action requests generated as a result of the QA audit are being processed in accordance with the corrective action program. Most of the identified corrective actions resulting from the evaluations of the findings have been completed.

Actions taken or planned include the following:

- Organizational changes and increased staffing levels have been implemented to improve the overall effectiveness of the program. The organizational changes permit improved management oversight of the process. An engineering position responsible for relief and safety valves has been established.
- A new procedure has been implemented that describes the IST program manager activities. This procedure includes requirements for performing periodic self assessments and program reviews, controlling changes to the IST manual, and providing interim changes to the manual.

- Changes to procedures have been made to address concerns with cold shutdown testing.
- The ten year review of the IST program is scheduled to be completed by December 31, 1997.

Improvements have been implemented to address previously identified weaknesses and believes that these improvements have adequately addressed the concerns.

V. *Conclusions*

In view of the foregoing information, PSE&G has reasonable assurance that the design bases is translated into operating, maintenance and testing procedures such that continued plant operation can be supported. The following conclusions are drawn with respect to the attributes identified in the "Overview" section.

- PSE&G has reasonable assurance that the necessary implementing procedures governing operations, maintenance and testing activities are in place. This element is dictated by regulatory requirements and was reviewed as part of the initial licensing effort and receives periodic review through the normal NRC inspection process and internal assessment activities.
- PSE&G has reasonable assurance that the implementing procedures are available for use and that changes to procedures are properly controlled. This is ensured through the procedure control process as described in the "Response to Question A" and by the effectiveness of the procedure control process as presented in this section of the response. In addition we employ an electronic Document Management System (DMS) that ensures controlled copies are maintained and readily available for a wide spectrum of facility documents including administrative and operating procedures, engineering documents and licensing documents.
- Based on the results of the internal and external assessments and the validation reviews, PSE&G has reasonable assurance that the design bases is reflected in the operating, maintenance and testing procedures such that continued plant operation is supported. PSE&G has reasonable assurance that the corrective action program coupled with the additional procedure review activities discussed in Attachment 2 will identify any existing discrepancies between procedures and the design bases and that appropriate actions will be taken in accordance with the Corrective Action Program and plant license if such discrepancies are discovered.

- PSE&G has reasonable assurance that the overarching processes related to: 1) translation of the design bases into operating, maintenance and testing procedures; 2) maintaining the plant configuration consistent with the design bases during implementation of operating, maintenance and testing procedures; and 3) ensuring proper implementation of operating, maintenance and testing procedures such that the plant configuration and equipment performance are maintained consistent with the design bases are considered adequate to support continued plant operation.
- Additional activities will be conducted to further assure that Hope Creek is configured and operated in accordance with the design bases, as discussed in Attachment 2.

HOPE CREEK GENERATING STATION

Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

RESPONSE TO QUESTION C

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

I. Overview

The purpose of this section is to provide the rationale for concluding system, structure, and component (SSC) configuration and performance are consistent with the design bases. This section will expand on the processes described in "Response to Question A" by identifying and demonstrating how key attributes of configuration control are fulfilled at Hope Creek Generating Station (HCGS), and thereby support the requested rationale. The key attributes of configuration management relevant to the requested rationale are that:

- the "as-built" configuration of plant SSCs is consistent with design bases requirements;
- the functional characteristics of plant SSCs are accepted through testing and surveillance activities; and
- processes to control installation and maintenance are defined and appropriately implemented.

The fidelity of the Hope Creek design bases was described above in the "Adequacy of Design Bases" section of this response. As discussed, there is reasonable assurance that the design bases were adequately developed and have been maintained over time. The discussion that follows concentrates on the translation of the design bases into the "as-built" configuration of the plant. Specifically, the remainder of this response analyzes the effectiveness of the above described configuration control attributes. Where appropriate, the discussion identifies known weaknesses and detailed actions taken and/or planned to resolve them, as follows:

- Implementation of the processes that maintain plant configuration and performance in accordance with the design bases is summarized in the "Process Controls" section.
- The "Validation Efforts" section describes actions which provide reasonable assurance that SSC configuration and performance processes are consistent

with the plant design bases and identifies potential process issues or improvements.

- The "Assessment of Effectiveness" section, first, summarizes key independent oversight findings related to the consistency of plant SSC configuration and performance with the design bases. Assessments discussed in this section have been performed by either internal oversight (Quality Assurance and Nuclear Safety Review), third party industry personnel, or the NRC. The section then assesses the effectiveness of process controls in ensuring consistent translation of the design bases into SSC configuration and performance with consideration given to validation reviews and independent assessment findings.
- "Improvements Implemented or Planned" section summarizes recent process or program improvements.
- "Conclusions" section provides the overall rationale for considering that HCGS SSC configuration and performance are consistent with the design bases.

II. Process Controls

The "Response to Question A" discusses in detail, the processes that maintain the design bases at HCGS. The primary process controls that ensure SSC configuration and performance are consistent with the design bases are discussed below.

NC.DE-AP.ZZ-0001(Q) "Design Bases/Input" (DEAP-1) procedure establishes a method for identifying design considerations and design input used in the preparation of design documents.

NC.DE-AP.ZZ-0002(Q) "Design Calculations and Analyses" (DEAP-2) procedure establishes the technical and administrative requirements for the development, maintenance, and control of design calculations.

NC.DE-AP.ZZ-0003(Q) "Modification Walkdown Program" (DEAP-3) procedure provides guidance to personnel participating in modification walkdowns. As part of the Design Change Package (DCP) process, walkdowns are performed to verify the existing configuration, identify plant and document discrepancies, and determine the feasibility, operability, maintainability, and testability of the proposed design or configuration change and ultimately, conformance to the approved Change Package (CP).

NC.DE-AP.ZZ-0004(Q) "Design Drawings" (DEAP-4) procedure outlines the process for preparing, reviewing and approving design drawings. New drawings or revisions to existing drawings are implemented by a CP developed according to the CP process. Drawings and pending impacts to drawings from CPs are available to site personnel electronically by the Document Management System (DMS).

NC.DE-AP.ZZ-0013(Q) "Processing Material, Equipment and Q-Listed Service Specifications" (DEAP-13) procedure defines the method for preparation, issuance and control of material, equipment, and Q-listed service specifications within the Nuclear Business Unit (NBU). Design considerations and design input information required by DEAP-1, along with the Configuration Baseline Document (CBD), and the SAR are reviewed to ensure design requirements and impacts are captured. This process ensures that that all material, equipment and Q-listed services are procured such that the design bases requirements are maintained.

NC.DE-AP.ZZ-0017(Q) "Modification Concerns and Resolution" (DEAP-17) process establishes a consistent method of documenting and resolving "minor" concerns encountered prior to and during the installation, testing and close-out of a CP.

NC.DE-AP.ZZ-0026(Q) "Engineering Evaluation" (DEAP-26) procedure applies to evaluations performed to document reviews, analyses, conclusions, or recommendations on topics including root cause analysis/problem analysis, engineering alternatives, safety concerns, or economic considerations. Evaluations that change the design bases for SSCs important to safety, or the basis of analyses, or conclusions stated in the SAR, receive a 10 CFR 50.59 applicability review or safety evaluation.

NC.DE-PS.ZZ-0034(Q) "Probabilistic Safety Assessment (PSA)" program is considered for use to assess the potential risk impacts.

HC.DE-AP.ZZ-0031(Q) "Component Functional Classification" procedure provides guidance in the proper classification of structures, systems, subsystems, components, parts, and associated work. Any modification, change or addition to a SSC is evaluated to determine whether it is required to perform a safety-related function. Potential failure modes of the SSC are considered in the process of the evaluation and are documented in a design classification sheet. The classification of SSCs is available to site personnel on Managed Maintenance Information System (MMIS).

Calculations prepared or revised independent of a DCP are reviewed for impact on station procedures and other design documents, and are reviewed for impact to the SAR. Refer to NC.NA-AP.ZZ-0059(Q) (NAP-59).

Calculations, drawings, Engineering Evaluations, component classifications, and equipment specifications are readily available through the site computer system.

These engineering process controls provide PSE&G with reasonable assurance that Hope Creek SSCs are operated and maintained consistent with the design bases.

III. Validation Efforts

Several assessments have been performed at Hope Creek to provide added assurance that SSCs are configured, constructed, and operated consistent with the design bases. They are described below.

A. Independent Design Verification Program (IDVP)

Before Hope Creek startup, PSE&G proposed a plan to the NRC staff for an IDVP to provide additional, independent assurance that design commitments and regulatory requirements were implemented into the plant configuration. The IDVP consisted of reviews of the design and the design process of representative Hope Creek SSCs in the High Pressure Coolant Injection (HPCI) system, Automatic Depressurization System (ADS), and Safety Auxiliary Cooling System (SACS).

The IDVP review was conducted in accordance with the *Program Plan for Hope Creek IDVP* (Revision 1), which was approved by the NRC staff on April 11, 1985. The IDVP was performed by Sargent & Lundy Engineers (S&L) beginning in April 1985. The IDVP was a comprehensive technical review, performed over a six month period by a team of qualified S&L personnel experienced in nuclear power plant design. The S&L team assessed the technical adequacy of the design of the selected systems, common as well as unique plant design features, and the design process used at Hope Creek. The purpose was to provide additional assurance that:

- FSAR commitments and regulatory requirements are being met;
- design process and interface control between technical disciplines ... functioning; and
- the design is technically accurate.

The S&L team contained a large number of multi-disciplinary engineers (mechanical, electrical, instrumentation and controls, and civil/structural). The team evaluated not only the design of the specific systems (HPCI, ADS, and SACS) but also the design of buildings containing the systems and common hazards and design considerations, such as internal flooding, missile protection, pipe whip, jet impingement, tornado missiles, etc. The team evaluated calculations, specifications, drawings, vendor information in order to draw conclusions relative to design accuracy, reasonableness and adequacy of design techniques, and the performance of the design process. The IDVP also included system and plant walkdowns.

The S&L team concluded that the IDVP and on-going design verification activities provided reasonable assurance that the overall design of Hope Creek was technically adequate and conformed to FSAR commitments and regulatory requirements. Conclusions were not limited to the three systems (and appropriate buildings) because the design sample was large enough and the review was in sufficient depth and detail to draw overall conclusions relative to effectiveness of the design process. Hence, results and conclusions from the sample were extrapolated to the overall Hope Creek configuration because the same effective design process which successfully implemented design commitments in the sample systems and buildings, was used throughout the entire power plant.

The NRC staff closely monitored the conduct of the IDVP. The NRC performed:

- A two-phase inspection of the IDVP implementation at S&L's offices in Chicago. The NRC staff's report of this inspection, 50-354/85-32, was provided to PSE&G in a letter dated July 8, 1985.
- An inspection of the documentation technically supporting the observation reports identified in the *S&L Final Report for the IDVP*, conducted at S&L's offices in Chicago. The NRC staff's report of this inspection, 50-354/85-54, was provided to PSE&G in a letter dated December 11, 1985.
- An inspection of agreed-on corrective actions identified in *the S&L Final Report for the IDVP*, conducted at Bechtel's offices in San Francisco in October 1985.

On the basis of the reviews and inspections conducted by the NRC staff to monitor the program, the staff concluded that the IDVP provided additional assurance that the design of Hope Creek meets the FSAR commitments and regulatory requirements.

Because Hope Creek was licensed in the mid-1980s, it is one of only a handful of late-vintage power plants (such as Limerick, Susquehanna, and Clinton) to have received an IDVP. The IDVPs were the most extensive, detailed, and comprehensive design and design implementation inspections imposed on any nuclear plants in the United States. Even the USNRC initiated Integrated Design Inspections (IDIs) which preceded the IDVPs, were performed by fewer individuals and involved smaller design samples. In addition, the IDVPs were closely monitored by an NRC team of engineering experts who subsequently concluded that the Hope Creek IDVP project plan was adequately implemented and that IDVP conclusions were supported by documented evidence. As such, IDVP results provide reasonable assurance in both the Hope Creek design and design implementation into the as-built configuration not available to the majority of power plants in the Country.

B. NRC Review of Hope Creek Technical Specifications (Report # 85-64)

During December 1985, the NRC conducted an inspection to determine whether the draft Hope Creek Technical Specifications (TS) and the FSAR agreed with the as-built configuration of plant SSCs, and to determine whether the draft TS requirements are definitively measurable or determinable. The inspection concentrated on plant SSCs having particular significance in minimizing the severity of potential accidents and accident consequences. The systems evaluated included: the reactor protection and safeguards actuation systems, standby liquid control system, primary and secondary containments and related support systems, emergency core cooling systems, and electrical power systems.

As part of this review, facility descriptions and operating characteristics for the SSCs found in the FSAR, the accompanying NRC Safety Evaluation Report (SER), and the "proof and review" (draft) TS were compared to licensee drawings, procedures, and actual plant hardware. The purpose of the review was to determine whether the as-built configuration of the SSCs was compatible with the safety analysis and the proposed TS.

The inspection found that the draft TS were compatible with the FSAR, the SER, the facility's procedures, and that the as-built plant was reflected by the engineering drawings, data, and in situ hardware. No violations were identified during this review. However, three exceptions were noted which required a follow-up inspection by Region I. The exceptions generally included unissued

TSs and TSs with incomplete data and, as such, were not significant detractors from the Hope Creek programs.

Thus, this too provides assurance that for the systems reviewed Hope Creek SSCs configuration and performance are consistent with the design bases.

C. NRC Special As-Built Team Inspection (Report # 85-58)

This inspection was conducted in December 1985 by NRC engineers to verify that selected systems were constructed substantially in conformance to the description contained in the FSAR and SER. The systems selected for inspection were those associated with meeting reactor safe shutdown and core cooling requirements. As such, the inspection included an examination of fluid systems, heating, ventilation and air conditioning (HVAC) systems, ac and dc power systems and instrumentation, and controls systems.

Independent dimensional measurements were made as part of extensive system walkdowns performed during the inspection. Various project specifications, drawings and design calculations were also reviewed during this inspection to verify that the SSCs were designed and fabricated such that they were capable of performing their intended functions as specified in the FSAR and whether the as-built configurations were in conformance with the FSAR, SER, and system specifications.

This as-built inspection focused particular attention in the following areas:

- Shutdown cooling functional systems between the Delaware River and the reactor core
- Systems and equipment necessary to fulfill the functional requirements of the Emergency Operating Procedures
- A compatibility check of the plant as-built condition with selected aspects of plant design specified in transient analysis portion of the FSAR

No violations were identified during the inspection. Four unresolved items which were identified in the areas of piping component and equipment supports, as well as instrumentation and controls. The inspectors determined that SSCs selected were designed and fabricated such that they were capable of performing their intended functions as specified in the FSAR.

D. Safety System Functional Reviews and System Functional Reviews

In 1985, the NRC initiated the Safety System Functional Inspection (SSFI) Program that entailed an in-depth investigation into the design adequacy and operational readiness of safety-related systems. The objective of an SSFI is to assess the operational readiness of selected safety systems by determining whether:

- The systems are capable of performing safety functions required by their design bases.
- Testing is adequate to demonstrate that the systems would perform all required safety functions.
- System maintenance (with emphasis on pumps and valves) is adequate to ensure system operability under postulated accident conditions.
- Operator and maintenance technician training are adequate to ensure proper operation and maintenance of the system.
- Human factors considerations relating to selected systems (for example, accessibility and labeling of valves) and the supporting procedures for those systems are adequate to ensure proper system operation under normal and accident conditions.
- Management controls, including procedures, are adequate to ensure that safety systems will fulfill the safety functions required by their design bases.

Starting in 1987, PSE&G initiated a series of Safety System Functional Reviews (SSFRs) and System Functional Reviews (SFRs). They are summarized below. The SSFRs and SFRs were conducted in a manner similar to the NRC's SSFIs. The review team first identified the design bases of the system. Concurrently, the team initiated the preparation of Technical Review Plans (TRPs) and data gathering activities. The TRPs served as guidance to the review team about the scope and detail of the review and highlighted potential problem areas identified by PSE&G. TRPs were prepared for each major area designated for review, including Maintenance, Operations, Design Change Process, Quality Assurance, Human Factors, and Training.

These efforts provide a historical perspective and were performed using the industry standards at the time of each review. It is recognized that subsequent to these efforts issues related to some of the systems reviewed have been raised. However the SSFR/SFR review efforts did evaluate aspects of the translation of the design bases information into the configuration and

performance of SSCs and provides reasonable assurance in assessing that the process is effective.

The inspection summaries below focus on design bases-related problems.

Station Service Water System (SSWS) SSFR

In September 1987, Cygna Energy Services and the Hope Creek Onsite Safety Review Group completed a detailed inspection of SSWS, its controls, and supporting systems. Observations related to design bases were: (1) the design bases was not easily retrievable; and (2) inconsistencies existed between licensing documents, engineering specifications, and surveillance test procedures.

Overall, this 1987 SSFR concluded that the configuration and performance of the SSWS met its design bases requirements under both normal and abnormal operating conditions.

Pneumatic Systems SFR

In May 1989, PSE&G and Cygna Energy Services completed a detailed inspection of station pneumatic systems including:

- Service Air
- Instrument Air
- Primary Containment Instrument Gas
- Emergency Diesel Starting/Control Air

One observation related to design bases was the effect of "loss of air" on valve position (3 valves).

Overall, in 1989, this SSFR concluded that configuration and performance of the pneumatic systems reviewed were consistent with their design bases requirements.

Safety Auxiliary Cooling System (SACS) SSFR

PSE&G, combined with United Energy Services and Gilbert Commonwealth, performed a detailed inspection of SACS starting in September 1990. The only observation related to design bases was that SACS was operated below the minimum design temperature (65°F) of the system. Subsequent engineering

calculations demonstrated that operating SACS as low as 32°F is acceptable. The UFSAR was subsequently revised to reflect this value.

Overall the results of this 1990 SSFR indicated that configuration and performance of SACS complied with its design bases requirements.

Primary Containment Isolation System (PCIS) SSFR

In June and July 1993, PSE&G and United Energy Services performed a detailed inspection of the PCIS and the Nuclear Steam Supply Shutoff System (NS4). The inspection did not note any significant observations related to design bases.

This 1993 SSFR determined the configuration and performance of PCIS and NS4 complied with their design bases requirements.

Residual Heat Removal (RHR) System SSFR

The PSE&G Safety Review Group completed a detailed inspection of the RHR system in June 1995. The team concluded that, except for the apparent inability to implement the fuel pool cooling assist mode as designed, the RHR system is capable of meeting its intended function. The fuel pool cooling assist mode could apparently not be implemented because the orifices required to implement fuel pool cooling assist were warped and useless. The SSFR determined the configuration and performance of the RHR system complied with their design bases requirements. The inability to implement the RHR mode of fuel pool cooling assist was not consequential at that time since other methods are available to provide for fuel cooling and the fuel pool cooling assist mode is only required after the 13th refueling outage, concurrent with a full core offload.

Radwaste Systems SFR

The PSE&G Safety Review Group completed a detailed inspection of the radwaste systems in August 1995. The Liquid Waste Management System (LWMS) was noted to perform below its design objectives (i.e., low efficiency). Observations included a number of instances in which equipment or procedures did not match the UFSAR. Suitable UFSAR change requests did not exist.

Overall, the team found that the Gaseous Waste Management System (GWMS) was capable of performing its functions and satisfying design requirements. The SSFR determined the configuration and performance of GWMS complied with their design bases requirements. The Process and Effluent Radiation Monitoring

System was noted to have an exceptional design with high sensitivity and reliability.

Although, LWMS processing efficiency is less than design objective, a review of the UFSAR determined the system is operating as defined. Additionally, LWMS is not relied on to mitigate an accident. A design review team is investigating the LWMS low efficiency problem.

Summary

The SSFRs and SFRs performed since 1987 did not identify any weakness that would have challenged system safety function. Problems or potential problems identified over the course of these inspections were dispositioned in accordance with the corrective action program.

In general, the SSFR/SFR was an effective tool in not only evaluating the design adequacy of a system, but also the adequacy of the various plant programs that affect the system such as maintenance, surveillance, and design change programs.

The collective results of these six SSFRs/SFRs tends to substantiate that the configuration and performance of SSCs reviewed are consistent with their design bases requirements.

E. Operational Readiness System Walkdowns

The Hope Creek Operational Readiness System Walkdown program was initiated to ensure the configuration and performance of each system was ready to support the safe and reliable startup and operation of the unit through the next cycle. The program consisted of a comprehensive detailed review, system failure history, and multi-discipline walkdown of components within the systems listed below:

- Station Service Water System
- Process Radiation Monitoring
- Emergency Diesel Generators
- Filtration, Recirculation & Ventilation System
- Control Rod Drive
- Reactor Recirculation

Walkdown guidance was provided by the *System Readiness Review Program* procedure (SC.TE-TD.ZZ-0023(Z)). General areas of focus included:

- Condition of snubbers, pipe supports, hangers and fasteners;
- Condition and placement of coatings and insulation;
- Evidence of corrosion;
- Unauthorized modifications, partial modifications or temporary modifications not in accordance with station programs;
- Surface condition of visible structural welds;
- Condition of barrier or penetration seals;
- Use of unauthorized chemicals;
- Evidence of bolt torque relaxation; and
- Evidence of discoloration on relays, cable insulation or electrical components.

In general, the majority of identified deficiencies were minor in nature, such as missing bolts, labeling, leaks, and cleanliness, none of which would have challenged system operation. One deficiency required corrective action prior to startup. Deficiencies were logged as Action Requests. These walkdown results, with the subsequent corrective actions, increased the overall level of reliability and performance of the above systems, and provide added assurance that the SSCs reviewed would have operated consistent with their design bases requirements.

F. Review of Selected Hope Creek Engineering Documents (SA-96-009) -- May 1996

In response to the back draft isolation damper issue (see Section IV.A below), a multi-disciplined PSE&G team assessed selected engineering work items to identify outstanding safety-related design bases issues that should be addressed and resolved by Station and Engineering Management. The work items which were selected from representative safe shutdown and risk significant systems, consisted of Justification of Continued Operation (JCOs), Engineering Evaluations (EEs), Discrepancy Evaluation Forms (DEFs), Design Change Requests (DCRs), and Design Change Packages. Document reviews and interviews were conducted with engineering and station personnel to determine if the documents identified any licensing commitment or design bases issues and if they were resolved or captured in the corrective action program.

A total of 361 documents were reviewed, covering 26 systems. The self-assessment noted that changes to the CBDs generated by DEFs were backlogged. As noted in section H below, the CBDs are scheduled for validation. The self-assessment determined the technical quality of EEs to be reasonable and sound.

Based on the depth of review, with no major findings, PSE&G believes that reasonable assurance exists that the engineering change processes are maintaining the configuration of SSCs consistent with their design bases.

G. Configuration Baseline Documentation Project

Principle Objectives

As discussed in the previous section entitled "Adequacy of Design" PSE&G undertook an extensive effort to compile design bases information, referred to as the Configuration Baseline Documentation project. The principle objective of the CBD project was to develop, consolidate, and document the specific design bases of the company's nuclear power stations. The focus of this effort identified and documented "why" the plant and supporting systems are designed and constructed to specific technical standards, safety guidelines, and/or specifications. As part of the process, the as-built configuration of the SSCs were compared to the design bases information.

Program Description and Scope

The CBDs were to be developed for each system or structure that was considered safety related, Technical Specification-related, or important to safety for Hope Creek Station. Twenty-nine CBDs were developed for Hope Creek. CBDs included the following elements: a functional description of the system; applicable codes and standards; regulatory documents; system operation; system and component design; accident analysis, as stated in the USFAR; and references. The CBDs were developed by outside contract organizations and were internally reviewed and accepted by system specific project teams. The development of a CBD required the following actions:

- Data retrieval (including Technical Specifications, UFSAR, licensing correspondence, and commitments);
- Indexing of data;
- Comparison of data to as-built SSCs (identify design inputs and licensing commitments, identify design outputs, comparison of design requirements

versus as-built condition, determine if all design input criteria have been met, identify discrepancies);

- Draft document preparation; and
- Consultant peer review.

Where discrepancies were identified between controlled documents, DEFs were generated. The DEF process included both operability and reportability reviews. Draft CBDs and associated DEFs were peer reviewed by the contract organization. The CBDs, DEFs, peer review comments, and comment resolutions were then submitted to PSE&G for review by the system-specific project team that included representatives from Design and System Engineering. Once comments were incorporated into the CBD to the satisfaction of the project team, the final CBD was reviewed by PSE&G and issued for use. A prioritization program, based on safety significance Probabilistic Risk Assessment (PRA) techniques, was used to categorize the DEFs for resolution schedules.

Programs and Topical Areas

As part of the CBD effort, PSE&G developed a series of programmatic standards that cover topical areas such as high energy line break (HELB), seismic design, environmental qualification (EQ), and fire protection. These standards were developed independently from the CBD program. The standards are referenced, where appropriate, within the CBDs to address program interfaces and requirements.

Verification and Validation Activities

PSE&G also instituted a process whereby a "vertical slice" review, known as a SSFR, was conducted on selected systems to review CBDs. As the CBDs were developed, many checks were made to ensure that the engineering data were consistent with the actual plant configuration. The configuration management program provided added assurance that NRC requirements were satisfied at Hope Creek.

A procedure was developed to control the maintenance of the CBDs. Any change to the CBD that resulted from a DCP was included as part of the DCP. If the DCP required design verification, the CBD change would also receive design verification as part of the DCP.

However, CBDs may also be impacted by processes apart from the design change process (i.e., revisions to calculations, engineering evaluations, and

review of Significant Operating Event Reports (SOERs)). To included such changes, the CBDs were revised and submitted to the Sponsor Engineer (system representative from Design Engineering) for review and approved by a Functional Supervisor in the Design Engineering organization. This type of text change to the CBDs was incorporated without design verification. Therefore, the CBDs could no longer be used as design input documents.

CBD Program Findings and Suspension

CBD documents were developed for 29 Hope Creek systems as of 1994. Specifically, in February 1994, PSE&G assessed the Hope Creek CBD to determine whether the CBDs:

1. had adequately met their stated goal to identify, categorize consolidate, and retain design bases information;
2. were of sufficient scope and depth;
3. should be continued for the remainder of the planned scope; and
4. could be developed in a cost-effective manner and yet achieve the same objectives.

The conclusion of this assessment was that the CBDs have generally met their goals as stated in the PSP and project specification DE-CB.ZZ-0001(Z). In addition, PSE&G managed to identify, categorize, consolidate, and retain design, licensing, and configuration bases in a readily retrievable format.

There were a few concerns identified in the assessment. One of the concerns was that several primary documents were inappropriately listed as input documents in the project specification. Furthermore, the amount of information contained in the CBDs was excessive and a large percentage of the information was readily available from other sources. Another concern identified was that the CBD maintenance program had some weaknesses that required improvement and attention.

A 1994 evaluation of the CBD DEF priorities and categories, revealed there were no DEFs issued requiring immediate action (evaluation with 7 days, resolution within 30 days) and that only three required near-term action (evaluation within 30 days, resolution with 90 days). The evaluation concluded that the design bases were adequate.

Current Status

In July 1996, a letter was distributed to Engineering and Licensing personnel discussing the proper application of the CBDs. The conclusion was that the CBDs should not be used as a design input source, but as a "road map" to the source documents. This approach was reinforced in a series of roll-out meetings conducted by the director of Design Engineering and Projects, as well as in the Engineering Support Program (ESP) training.

H. Configuration Baseline Document Validation

The CBD validation effort was initiated in late 1996 to instill confidence that the design information contained within the CBDs accurately reflect as-built and operating conditions at Hope Creek. As part of this effort, PSE&G is reviewing the Hope Creek operating procedures to ensure that key operating parameters meet their design bases. A validated CBD integrates all the design information to form one congruent design picture ensuring safe and reliable operation of the plant and maintaining compliance with NRC regulations.

The CBD validation of the systems described below, verifies the design bases information contained in the CBDs by ensuring consistency with PSE&G design output and operating documents such as the UFSAR, Technical Specifications, P&IDs, operating and emergency procedures, test procedures, I&C program, and ASME Section XI program. The assessments used design and licensing bases document reviews, and personnel interviews to identify weaknesses, missing information, or disconnects in the design bases documentation, operating procedures, and/or engineering design output documents. As appropriate, the validations focus on mechanical systems design, electrical systems design, operations, surveillance and testing and/or instrumentation and control design. CBD validations were performed on the following systems:

Station Service Water System Self-Assessment (SA-96-012)

This CBD validation was performed on the SWSS and the Service Water Traveling Screens in April and May 1996. The SWSS Chlorination system was not reviewed.

The following strengths were identified:

- Prior to recent modifications to the systems, the CBD was adequately maintained
- The hydraulic model for the system is thorough and benchmarked.

Weakness noted were: (1) Inconsistencies between procedures and design calculations; and (2) A detailed single failure analysis for service water system could not be located. One finding involved the original design calculation for the overboard discharge line that contained errors which resulted in a conclusion that there would be insufficient flow through the overboard path to ensure that SACS heat exchanger outlet temperatures remained below the 95°F design limit following a Safe Shutdown Earthquake (SSE) with concurrent Loss of Offsite Power (LOP), the UHS at its TS temperature limit, and TS minimum river level. An administrative TS temperature limit was instituted until a full analysis of system is completed. Deficiencies noted are tracked by the corrective action process. In NRC inspection report 50-354/96-05, the inspector noted that PSE&G was taking appropriate corrective action to ensure the TS temperature limit is promptly restored. This review along with engineering efforts to correct identified problems, provide reasonable assurance that from this point onward, configuration and performance of the system will be maintained consistent with its design bases requirements.

Safety & Turbine Auxiliaries Cooling System (STACS) Collegial Self-Assessment (SA-96-015)

In June through August 1996, CBD validation was performed on the STACS. The non-safety related Turbine Auxiliaries Cooling (TACS) portion of the STACS system was not reviewed.

The following weaknesses were identified: (1) original plant design calculations did not accurately reflect the as-built configuration, configurations allowed by operating procedures, or the appropriate effects of a loss of instrument air following a Loss of Coolant Accident/Loss of Offsite Power; and (2) maintenance of design bases documentation, such as design calculations and CBDs, appeared to be weak.

Findings included concerns relative to the bases for TS allowable out-of-service times for inoperable EDGs, SSWS pumps, and SACS pumps and identification of inadequacies in the SACS startup testing. Deficiencies were logged and tracked by the corrective action process, closure of these items is in progress. Although the validation identified some weaknesses in the design bases, administrative controls and plant modifications were implemented to ensure the plant will continue to operate within its design bases. This validation along with additional engineering efforts provide reasonable assurance that configuration and performance of STACS is consistent with its design bases requirements.

HOPE CREEK GENERATING STATION
Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

Emergency Diesel Generators (EDG) Self-Assessment (SA-96-018)

This inspection is in progress. A draft report for the mechanical portion of the system noted that comprehensive reviews performed in the past (EDSFI and QA EDG Review) identified previous design weaknesses. The inspection identified several minor discrepancies between the FSAR and CBD. Additionally, several setpoint calculations did not incorporate instrument inaccuracies. No operability problems were identified.

High Pressure Coolant Injection Self-Assessment

This inspection is in progress. To date no operability problems were identified. The team has not identified any issue which would dispute that the HPCI CBD accurately reflects the as-built design and operating conditions.

CBD Validation Summary

In summary, the CBD validations performed to date have identified some weaknesses in the design bases. These weaknesses are being aggressively addressed by the engineering staff and tracked through the corrective action process. A validated CBD ensures that a strong link exists between plant and engineering organizations, that essential knowledge is not lost, and that the CBDs accurately reflect the system design bases information. These CBD validations along with the corrective action processes provide reasonable assurance that the configuration and performance of Hope Creek SSCs are consistent with the design bases.

I. Service Water System Operational Performance Inspection (SWSOPI)

This is a PSE&G sponsored inspection consisting of a detailed review of the configuration and performance of the Hope Creek SSWS and SACS to ensure they are consistent with the design bases. The inspection includes plant document reviews to ensure consistent application of design bases down through the design output and operating documents, such as the UFSAR, Technical Specifications, P&IDs, elementary and logic diagrams, operating and emergency procedures, test procedures, ASME Section XI and other programs, and CBDs. The inspection also places a heavy emphasis on a review of the response to and implementation of Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*.

The SSWS portion of the SWSOPI is complete and the SACS portion of the inspection is on hold.

Based on the review to date, the team concluded the detail in SSWS analyses reviewed is good, root cause analyses conducted recently were strong efforts and extensive. Engineering self-assessments on SSWS identified many of the issues found during the SWSOPI. Where necessary Action Requests were generated to track deficiencies.

The SWSOPI also identified the following problems:

- internal flooding and single-failure issues on the RACS service water discharge line,
- internal flooding issues for breaks in the non-seismic effluent discharge piping lines in the service water intake structure,
- weakness in the implementation to conduct single failure review of the SSWS, and
- weakness in revision control of a design calculation for SSWS.

The SWSOPI concluded recent engineering efforts provide assurance that the configuration and performance of SSCs reviewed were consistent with their design bases given the current administrative limits on Ultimate Heat Sink (UHS) temperature and tide level (see Section III.H above). The team also considered noteworthy both the efforts and magnitude of the corrective actions that have been self-identified to improve performance and reliability of the SSWS. The SWSOPI, including the SACS portion, is currently scheduled to restart in March 1997.

IV. Assessment of Effectiveness

A. Independent Assessments

Since construction, Hope Creek has been the subject of several extensive internal and external assessments that have, in part, evaluated the consistency between plant configuration and performance and the design bases. Specifically, a portion of each assessment has provided the plant staff with assurance that the unit is being operated and maintained in accordance with the design bases including the UFSAR and SER. They also have reviewed the effectiveness of the configuration control process at Hope Creek. These assessments are summarized below.

NRC Readiness Assessment Team Inspection (IR 96-80)

Management meetings were held between PSE&G and the NRC at the beginning and halfway through RFO6, in late 1995 and early 1996, to discuss improvement efforts. These efforts were set forth in the *Hope Creek Outage Completion Plan* (OCP), which was approved by the NRC in January 1996.

The NRC performed a Readiness Assessment Team Inspection (RATI) prior to startup from RFO6. The RATI reviewed the Impact Plan and the OCP to develop screening criteria for the selection of appropriate areas for inspection. The team assessed plant performance in the areas of Management Programs; Independent Oversight and Self-Assessment; Operations; Maintenance and Planning; and Engineering.

The inspectors identified two issues related to design bases that required licensee resolution before plant restart. The first design bases issue, which arose during a review of DCPs, identified improper installation of reactor building ventilation backdraft isolation dampers (BDID). The root cause investigation determined that the BDIDs were installed incorrectly during construction. The deficiency was noted again in 1992, and that the orientation of the main steam tunnel BDIDs were then corrected. The other dampers were determined to be less risk significant and were not corrected at the time. An August 1992 engineering evaluation concluded that the ability to safely attain and maintain plant shutdown was not compromised by the misorientated dampers.

As part of the actions taken to identify the extent of this deficiency, an engineering self-assessment, of over 360 Hope Creek EEs, DEFs, JCOs and DCRs was performed (discussed in, Section III.F above). The result of the review was that no new items added to RFO6, however 20 Business Process (BP) Action Requests were initiated to enhance, clarify or improve the design bases documentation. This self-assessment provides assurance that known discrepancies were addressed appropriately and this deficiency appears to be an isolated occurrence. Other efforts combined with these reviews provide reasonable assurance in the adequacy of our design bases.

Seven other DCPs were reviewed by the NRC to evaluate the extent to which they addressed the adequacy of the installation instructions, appropriateness of retests, procedural update requirements, and conformance to the licensing and design bases as documented in the UFSAR. The DCPs reviewed were generally of good quality with sufficient documentation to permit evaluation of the effect on design and licensing basis, appropriate retest instructions, necessary procedure and UFSAR change requests, and adequate installation instructions.

These findings underscore the effectiveness of the configuration control process at Hope Creek which, in turn ensures plant configuration and operation are consistent with the design bases

The second design bases issue involved a conflict between battery temperatures in the UFSAR, TS, and battery sizing calculation for the 125 and 250 volt 1E batteries. The class 1E battery sizing calculations used the lowest expected temperature of 72°F. TS requires electrolyte temperature to be above 60°F. PSE&G took action to revise the surveillance procedures to notify the supervisor if electrolyte temperature dropped below 72°F, while an engineering evaluation was performed. Engineering subsequently determined the battery had sufficient capability to meet its design bases load at a temperature of 60°F. Based on these actions, the team did not have a restart concern.

The NRC determined that management, staff, procedures, programs and processes were adequate to support safe restart and continued operation. The NRC also identified that PSE&G demonstrated adequate processes for configuration control of plant systems, noting no deficiencies in configuration control were identified during plant walkdowns of several systems important to safety. Further, based on these findings, the NRC determined that PSE&G had adequate measures in place to ensure issues that needed to be addressed before unit restart were identified and addressed and that adequate corrective action had been taken to address previously known problems.

Therefore, this inspection provided added assurance that (1) the adequacy of the configuration control process; and (2) that the plant is operating in accordance with its design bases.

NRC Electrical Distribution System Functional Inspection (EDSFI)

The NRC conducted an EDSFI at Hope Creek from January 27 through February 14, 1992. The purpose of this inspection was to determine whether the electrical distribution system (EDS) was capable of performing its intended safety functions as designed, installed, and configured. The team also assessed PSE&G's engineering and technical support of EDS activities. The team performed plant walkdowns and technical reviews of studies, calculations and design drawings pertaining to the EDS and conducted interviews of corporate and plant personnel. The review covered portions of the onsite and offsite power sources. The review included walkdowns of equipment to verify as-installed condition, maintenance, calibration and testing. Items covered included the emergency diesel generators, power transformers, fast and slow transfer scheme, circuit breakers, batteries, and battery chargers.

The inspection team concluded that the EDS is capable of performing its intended function, and the engineering organization provides adequate engineering support for the safe operation of the plant. The inspection found that three of the activities appeared to deviate from the Hope Creek UFSAR, which resulted in a Notice of Deviation. The three deviations were related to the Emergency Diesel Generators. These deviations were entered in the corrective action system and subsequently closed out.

The EDSFI and corrective actions, which were implemented, provided reasonable assurance that the configuration and performance of electrical distribution system was consistent with the design bases requirements.

B. Summary

PSE&G believes that these independent assessments, performed by the NRC, provide reasonable assurance that Hope Creek has maintained configuration and performance consistent with the design bases.

V. Improvements Implemented or Planned

The Nuclear Engineering Department recently implemented or reinforced several actions to provide additional assurance that configuration and performance of SSCs are maintained in accordance the design bases at Hope Creek.

- Require a line-by-line, detailed review of the DCP by the Peer Reviewer;
- Schedule the appropriate time for the Peer Reviewer to perform the required detailed review;
- Document deviations from standard practices, vendor recommendations and document acceptability of specified equipment;
- Require Peer Review by Installation and Test Group for testing sections of the DCP;
- Require conceptual and constructability walkdowns that are documented in design analysis;
- Require Installation and Test Engineering to walkdown the modification prior to installation;
- Include Station Maintenance, Operations, Station Planning and the DCP Installer to review the change package;
- Implement a training and qualification program for Modification Engineers; and

- Perform third party reviews.

Additionally, Design Change Process Review effort is part of overall process improvement. The scope of this effort is to benchmark the process, and recommend procedure changes to improve the program. The process reviews are underway, and recommendations to streamline will be incorporated in 1997.

VI. Conclusion

In view of the foregoing information, PSE&G has reasonable assurance that SSC configuration and performance are consistent with the Hope Creek Generating Station design bases because:

- Plant configuration was originally licensed by the NRC consistent to the design bases;
- Processes have been improved that will ensure consistency with and conformance to the design bases;
- The processes are effective, while the results of performance-based assessments and readiness programs indicate deficiencies remain to be corrected. These deficiencies are being aggressively addressed by the engineering staff and tracked through the corrective action process. The primary assessments or programs that support this conclusion, completed in the last 18 months, are:
    1. Service Water Operational Performance Inspection,
    2. Residual Heat Removal System Safety System Functional Review,
    3. Radwaste System Functional Review ,
    4. Operational Readiness Walkdowns,
    5. Configuration Baseline Document Validation, and
    6. Review of Selected Hope Creek Engineering Documents.
- The results of the following NRC assessments also demonstrate that SSC configuration and performance are consistent with the design bases.
    1. Readiness Assessment Team Inspection (1996), and
    2. Electrical Distribution System Functional Inspection (1992).
- The Hope Creek Corrective Action Program, along with other configuration control processes, as described in the "Response to Question D",

demonstrated to be effective during the above-mentioned assessments or programs in maintaining configuration in accordance with the design bases.

In addition, to give added assurance in this area, beyond the reasonable assurance that exists at Hope Creek, activities will be conducted to provide additional assurance that the plant is configured and operated in accordance with the design bases, as discussed in Attachment 2 of this response.

RESPONSE TO QUESTION D

Provide information on the 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.

I. Corrective Action Program

A. Hope Creek Action Request Process

The Hope Creek site uses the Action Request Process to collect, track and control all types of problems for resolution. This process encompasses nonconformances in plant design and problems in plant design documents. For conditions adverse to quality, the Action Request Process provides for evaluation of the problem. Resolution of the problem and implementation of any corrective actions is contained in the Corrective Action Program part of the process. Problems can be identified from numerous sources, both internal and external. This process allows all employees and contractors the ability to enter any problems for evaluation and response.

This section provides a brief overview of the operation of the Action Request Process and the Corrective Action Program including conditions adverse to quality, operability determinations, reportability determinations and action closure. A summary of how the Operating Experience Program adds external information to the process is included. The important role of the Employee Concerns Program for identifying internal issues for entry into the Action Request Process is also described. In addition, in Section V, information is provided on the effectiveness of PSE&G's implementation of these programs.

The process has evolved substantially from the spring of 1995, after various program weaknesses were identified by both internal and external assessments. These program weaknesses included: reporting threshold was too high, root cause analyses did not get to the root cause or failed to correct the root cause, there was insufficient follow through on corrective actions, root cause/cause determinations and corrective actions were not completed in a timely manner, and corrective actions were not always effective at correcting problems. To correct these weaknesses, a new process was designed and new governing procedures were implemented. A root cause manual was developed and training was provided for root cause investigators. Section C contains recent

inspection results on the Corrective Action Program. These inspections generally discuss the improved process as providing the necessary capabilities.

The Corrective Action Program utilizes evaluation managers to lead the evaluation of problems and to assign tasks for resolution as appropriate. The program is designed such that the responsible department manager retains ownership and accountability for resolution of the issues. Operability impact and reportability determinations are made within the evaluation process. The evaluation manager is responsible for assuring adequate and timely resolution of the problem.

Conditions adverse to quality require causal analysis to help ensure effective countermeasures are implemented. The type of causal analysis depends on the significance level of the condition adverse to quality (Level 3 conditions require trending rather than causal analysis.). Recurrences of conditions adverse to quality result in higher action levels within the Corrective Action Program.

Those problems requiring design or configuration changes for resolution are tracked by Hope Creek site engineering in the Design/Configuration Change Tracking System. This system tracks creation of the design change package, implementation and closure [described in "Response to Question A"]. All design changes require 10CFR50.59 reviews and updating of the UFSAR as applicable.

B. Action Request Process Operation

This Action Request (AR) Process description applies to the Nuclear Business Unit (NBU) which encompasses both Salem and Hope Creek Generating Stations. The AR Process provides the method for reporting and resolving conditions adverse to quality as defined in 10CFR50, Appendix B, Criteria XVI, and business process type enhancements, thus providing a single point of entry for items requiring corrective action. Coding is provided to separate conditions adverse to quality from business process items in the following manner.

- Corrective Maintenance (CM) action requests are used for reporting equipment related conditions adverse to quality.
- Condition Resolution (CR) action requests are used for reporting programmatic conditions adverse to quality other than hardware (i.e. administrative control, procedural or human error events).
- Business Process (BP) action requests are used to report business process enhancements, requests for information or support, etc..

The AR Process is described in procedure NC.NA-AP.ZZ-0000(Q), "Action Request Process". This procedure provides the directions necessary for conditions requiring immediate action, initiation, review, approval, management review, and closure. The three AR types are controlled by the following procedures:

- Corrective Maintenance (CM) type: NC.NA-AP.ZZ-0009(Q), "Work Control Process"
- Condition Resolution (CR) type: NC.NA-AP.ZZ-0006(Q), "Corrective Action Program"
- Business Process (BP) type: NC.NA-BP.ZZ-0000(Z), "Business Processing of Action Requests"

During the initiation phase of the AR process, individuals document the condition by providing a description of the condition. In addition, the process calls for an impact statement, identification of regulatory reporting requirements that may apply, suspected cause or source of condition, corrective actions previously taken, recommended corrective action and recommended evaluation manager. Significance levels are determined by the initiator and reassessed by reviewers and approvers to establish the importance of the action.

While the major source of actions identified come from employees and contractors during the normal course of their day-to-day work, other programs such as the Operating Experience Program, 10CFR Part 21 reports and the Employee Concerns Program provide actions to be processed in the AR Process.

C. Conditions Adverse to Quality

The Corrective Action Program [NC.NA-AP.ZZ-0006(Q)] provides the instructions and guidance to ensure conditions adverse to quality are dispositioned and corrected in compliance with 10CFR50, Appendix B, Criterion XV, (Nonconforming Materials, Parts, or Components) and Criterion XVI, (Corrective Action). The program applies to Corrective Maintenance (CM) and Condition Resolution (CR) type Action Requests initiated in accordance with (IAW) NC.NA-AP.ZZ-0000(Q) Action Request Process. Evaluations are required by procedure and are conducted to determine the extent of the condition and to determine corrective actions as follows:

Significance level 1 - A root cause investigation and a root cause analysis report (RCAR) is completed in accordance with DTG-CAP-003, "Root Cause Manual". The RCAR is expected to include a detailed root cause and causal factors analysis, recommended corrective actions, and follow-up actions to verify the effectiveness of the corrective actions.

Significance level 2 - An apparent cause evaluation is conducted in accordance with the Root Cause Manual. The evaluation is expected to include an apparent cause and causal factors analysis, and any corrective actions to correct the condition.

Significance level 3 - The Evaluation Manager may either specify and initiate actions to correct the condition, document the evaluation, or assign trend code(s) and close the action if no open tasks remain.

Significance level 4 - These tasks have been evaluated to not be conditions adverse to quality and are no longer a level in the Corrective Action Program. They are now in the BP or Business Process part of the Action Request Process.

Trending of problems and the implementation of the Maintenance Rule requirements have been incorporated into the Corrective Action Program.

D. Operability Determinations

The reviewer of an AR is expected to immediately notify the SRO approver if an operability concern exists. The SRO approver is then responsible to immediately notify the on-duty SNSS/NSS if a condition adverse to quality exists requiring an operability determination. Operability determinations are performed in accordance with Operability Determination procedures and documented. If required, engineering support may be requested to perform an in-depth follow-up assessment of operability and provide documentation.

Operations Management presents Significance Level 1, 2 and 3 conditions requiring follow-up operability assessment to the station management team for consideration of the following:

- Designation of the Evaluation Manager or Responsible Department.
- Concurrence with or change to the significance level.
- Designation of any short term actions and requirements.

- Identification of any special considerations.
- Authorization of a team investigation for significance level 1 conditions or other conditions as deemed appropriate.

Upon being informed of an operability issue, the on-duty SNSS/NSS determines and directs any immediate actions needed to place the plant in a safe condition. The results of the initial operability screening for the affected Structure, System, or Component (SSC) are documented in accordance with Operations Department procedures.

The operability determination process is the element of the Action Request process whereby adverse conditions are evaluated for impact on the plant design bases relative to the ability of SSC's to perform their intended safety function. This provides a mechanism for the Operations Departments to maintain cognizance of adverse conditions with the potential to impact plant safety, and thereby implement any restrictions on plant operation or compensatory actions to maintain the plant in a safe condition.

E. Reportability 10CFR50.72 & 10CFR50.73

The Senior Reactor Operator (SRO) approver has the responsibility to determine the effect of conditions adverse to quality on structures, systems, and components (SSC). If the condition identified requires 10CFR50.72 notification, the SRO approver immediately contacts the on-duty Senior Nuclear Shift Supervisor/Nuclear Shift Supervisor (SNSS/NSS). Licensing is contacted for assistance in determining reportability.

The Nuclear Licensing and Regulation Department reviews the reportability of conditions adverse to quality according to state and federal regulations. When a 10CFR50.73 reportable event is identified, Licensing notifies the evaluation manager responsible for resolving the condition and obtains information for required report preparation. If it is determined that a notification was not made within the required time, the licensing representative would contact the SNSS/NSS to make the required notification.

F. Action Closure

Corrective actions are tracked to completion/implementation and closure actions are documented by the assigned Evaluation Manager. The organization assigned the implementation of the associated corrective action(s) is required to maintain the activity in the "active status" until the corrective action is implemented and documentation of completion is provided.

When all evaluations, corrective actions, verifications, and all other activities are implemented and completed, the Evaluation Manager verifies that actions specified in the evaluation have been implemented and the condition has either been resolved or new activities are created to correct the remaining conditions, as required. The Evaluation Manager assures codes are assigned according to the Root Cause Manual to assist with the identification of adverse trends. When all activities are implemented and documented as "complete," the Evaluation Manager then places the action in the approved status. The Evaluation Manager also ensures hard copy documentation is forwarded for appropriate record keeping.

G. Training and Responsibility Identification

General Employee Training for access to the Hope Creek Site includes a section on the Corrective Action Program. The training identifies the responsibility of every employee to identify conditions adverse to quality and to report these conditions through the Action Request Process. The electronic entry and manual form method of entry are described. Also, the process of contacting the Senior Nuclear Shift Supervisor for operability issues is described. Employees are also told that additional information and training is available from the Corrective Action Group. Training is available from the Corrective Action Group upon request.

The NBU Work Standards Handbook also reinforces the Corrective Action Program by stating that every employee is responsible and has the authority to identify conditions adverse to quality and to report these conditions through the Corrective Action Program. Management expectations are constantly reinforced during indicator review meetings and during problem identification discussions.

II. Operating Experience Program

The Hope Creek Operating Experience Program is used to analyze industry operating experience in order to implement timely actions to prevent or reduce the consequences of similar occurrences at this site. [Procedure NC.NA-AP.ZZ-0054 (Q), "Operating Experience (OE) Program"]

Documents containing operating experience information from outside the Hope Creek site are screened for applicability to the Hope Creek site by the OE Program. The following documents are included in the screening:

1. INPO Significant Operating Experience Reports(SOERs)

2. INPO Significant Event Reports (SERs)

3. INPO Significant Event Notifications (SENs)

4. INPO Significant by Others (SOs)

5. INPO Operations and Maintenance Reminders (O&MRs)

6. NRC Bulletins and NRC Generic Letters

   These important documents within the OE Program have provided information on design problems. Many of these documents have resulted in the implementation of design improvements at Hope Creek. Most of these documents have required responses back to the NRC describing the design improvement actions taken.

7. NRC Information Notices

8. Vendor Reports (including 10CFR Part 21 reports)

9. Industry Operating Experience Reports (OEs)

10. Industry Plant Status Reports (PSs)

11. Any other documents containing information appropriate to the OE Program

12. Nuclear Plant Reliability Data System (NPRDS)

Problems identified as applicable to the Hope Creek site are documented and entered into the Action Request process for resolution. The corrective action database is then used to track the status of evaluations and corrective actions.

OE Program effectiveness is monitored by means of indicators. Periodic self-assessments are conducted to assure continued program effectiveness and improvements.

III. 10CFR PART 21 Process

Conditions adverse to quality that are provided to PSE&G under 10CFR21 and conditions adverse to quality that may require reporting by PSE&G under 10CFR21 are entered into the Corrective Action Program. The Corrective Action Program provides the process to determine operability impact and reportability requirements. The Corrective Action Program tracks the issue until the necessary corrective actions are completed.

The conditions adverse to quality provided to PSE&G under 10CFR21.21(b) are received by our Materials Center and forwarded to the Operating Experience Program for action assignment in the Corrective Action Program. Vendor documents that could result in a 10CFR21 reporting requirement for PSE&G are also entered into the Corrective Action Program by the Operating Experience Program to assign evaluation responsibilities. Conditions adverse to quality, identified at PSE&G, that have potential reporting responsibility under 10CFR21 are entered into the Corrective Action Program by the identifying department. Those conditions adverse to quality that could have reporting requirements under 10CFR21 have to have Form 1 of procedure NC.NA-AP.ZZ-0035(Q), Nuclear Licensing and Reporting completed, to provide establishment of reporting requirements.

IV. Employee Concerns Program

The Hope Creek site employee concerns program can be a source for design bases issues and configuration control issues. When such issues arise, and evaluations have to be performed to determine the need for corrective actions, the actions are entered into the Action Request Process. The employee concerns program provides another way of assuring design information and configuration control is maintained.

V. Effectiveness

A. Program Requirements

The Corrective Action Program at Hope Creek has programmatic requirements for ensuring that significant conditions adverse to quality are corrected and that those corrective measures preclude recurrence. The program requires that an effectiveness review be performed for any significance level 1 condition adverse to quality after all evaluations, activities, and corrective actions are complete. Effectiveness review plans are required to be presented to the Corrective Action Review Board (CARB) during the root cause presentation for all significance

level 1 ARs and are subsequently required to return to CARB for validation of effectiveness results prior to the issue being closed. An effectiveness review is expected to assure the following:

1. All corrective actions are complete.
2. The condition and causes(s) were corrected and have remained corrected.
3. That no additional actions are required.
4. That the specified corrective actions did not create any new conditions.

Corrective Action Program effectiveness is ensured and monitored for all Condition Resolutions (CRs) by the following:

1. Program Performance Indicators
2. Evaluation Manager's review of completed activities
3. Corrective Action Review Board for root cause evaluations (significance level 1 issues)
4. Corrective Action Review Committee for apparent cause evaluations (significance level 2 issues)
5. Program Internal Self Assessments
6. Internal Audit Programs

Corrective Action Program performance indicators include components in the areas of timeliness, schedule adherence, and quality. Timeliness indicators include average age, average time to complete evaluations, average time to complete tasks, and total actions open. Schedule adherence tracks task completion to the schedule. Quality assessments are done by the Corrective Action Review Board and Corrective Action Review Committee on the level 1 and level 2 causal evaluations.

B. Results

The cur.ent Corrective Action Program and the Action Tracking System (previous corrective action program) contain numerous examples of design and design bases issues that were correctly addressed and the design documentation updated including the UFSAR. Both tracking systems require all actions to be completed before an issue is closed. The following table shows examples of issues which were evaluating design bases questions that resulted in the submittal of LERs and have either design bases corrections or design changes in progress or completed:

DESIGN BASES REVIEWS RESULTING IN LERs

LER NUMBER	DATE	ISSUE	DESCRIPTION
354/96-015	4/10/96	A preliminary hydraulic calculation indicates that the minimum flow required to maintain the SACS HX outlet temperature below the 95 degree design limit may not be available following a safe shutdown earthquake.	A design and configuration baseline documentation validation of the service water system is complete.
354/96-009	3/17/96	As found positions of the service water to SACS HX throttle valves were inconsistent with assumptions used in calculations for SSWS/SACS operation.	The design change for SSWS backwash strainers and SSWS flow balancing were completed. Throttle valves were positioned to ensure adequate SSWS/SACS performance under all design bases conditions.

These examples of previously identified problems and the implemented corrective actions provide confidence that our corrective action process is effective in meeting Appendix B criteria.

C. Inspection Results

This section contains some of the more significant statements from some of the recent Hope Creek inspections and reports from the NRC and our QA Department regarding the Corrective Action Program (CAP) and related areas. These inspections indicate that we do have the appropriate process installed. Some of these inspections did identify implementation weaknesses which are being addressed under the response effort to the individual inspections. However, the inspections do show that the overall process effectiveness is sound.

Readiness Assessment Team Inspection (RATI 50-354/96-80) Dated April 24, 1996

Section 3.2.4 Corrective Action Program

"d. Conclusions: The CAP was sufficiently established to identify and resolve plant deficiencies in a timely manner and was functioning acceptably. Personnel understood how to use the program and management expectations for its use. The program provided sufficient tracking of required actions for identified problems. Appropriate requirements were in place for the classification and timeliness of resolutions."

"The team noted several minor issues with the program regarding: lack of feedback to the initiators and supervisors, differing significance level guidance, and misunderstanding of the 30 day evaluation completion time. The team did not consider these significant from restart viewpoint since the adequacy of corrective actions were not adversely affected."

" Based upon the review of performance indicators it appeared that the overall completion of CREVs, completion of CRCAs and quality of significance level 1 evaluations/corrective actions have improved and met management's expectations."

"The initial development of trending data of corrective actions has recently provided additional insight to the management oversight and levelized the expectations for evaluation quality and content."

HOPE CREEK GENERATING STATION

Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

Quality Assessment Department Audit Report 96-190-2, Corrective Action Audit - Performed October 30 through November 26,1996

Executive Summary

"Effectiveness Statement: The audit team determined that the Corrective Action Program is adequate and supportive of the continued operation of Hope Creek and Salem operating mode changes. Areas of improved performance were noted; however, the audit identified ten (10) areas of deficiencies that will require further refinement and or management attention to improve performance. Increased emphasis for the implementation of the program is needed particularly for the performance of the evaluation manager's role and responsibilities to ensure appropriate corrective action items are identified and implemented satisfactorily."

"Improved Performance: The audit team continued to see improvement in problem identification as evidenced by the continuing number of action requests initiated. The QA corrective action cultural survey continues to demonstrate an increased understanding of when and how to initiate an action request and management support for the process."

"The use of dedicated corrective action groups within the station's departments continues to demonstrate improved focus for implementation of the process and timeliness."

"The Corrective Action Program's control of documentation and transmittal for record retention has demonstrated improvement. The audit identified one instance where the Root Cause Analysis Report was not transmitted to the Corrective Action Group (CAG)."

"During the last QA Corrective Action Audit it was identified through a Maintenance Surveillance and the survey that people were reluctant to initiate Action Requests for issues related to human performance. This audit's survey attempted to clarify this issue through responses from the following:

- People are reluctant to initiate ARs.
- View themselves as being reluctant to initiate ARs.

- Aware of instances where people were reluctant to initiate ARs within the last three months.
- Knowledge of what actions were taken even though the individuals may have been reluctant to initiate the AR."

"The survey's responses demonstrated that the reluctance issue reported during the last QA audit was perceived rather than factual."

NRC Integrated Inspection Report 50-354/96-09 - Dated December 5,1996
Section 07.2 Corrective Action Program

The reports conclusions state that the "licensee's corrective action program procedures and process as described appear to be effective in the identification and resolution of conditions adverse to quality. With several minor exceptions, the process was in use site-wide and received strong management support. However, inconsistent understanding of AR initiation threshold, lack of feedback on ARs submitted, and potentially inadequate training on the AR process indicated that management's expectations regarding AR usage were not always being met or not clearly communicated."

"The performance indicators provide a solid method for management to remain aware of the status of areas monitored."

"The corrective action program results have been mixed based upon performance indicators, QA audit results and continued and repetitive deficiencies which challenge operators still exist."

VI. Discussion Of Recent Findings

While we believe reasonable assurance has been provided in "Response to Questions B and C", PSE&G recognizes that deficiencies in design bases consistency within the Hope Creek plant configuration and procedures have recently been discovered. Some examples of recent findings are listed below. There was an unplanned release of radiological material from the South Plant Vent on April 5, 1995. RHR shutdown cooling was bypassed in July 1995. The RATI team identified the improper installation of several high energy line break backdraft isolation dampers in the Filtration, Recirculation, and Ventilation System (FRVS) in early 1996. Hope Creek is also continuing to investigate the Emergency Core Cooling System suction strainer issue. This is an on-going effort to address the concerns of NRC Bulletin 96-03.

The issues described above are being aggressively addressed by the engineering staff and tracked through the corrective action process. The goal is to minimize these types of occurrences. However, if a problem does occur, it will be dealt with in accordance with Hope Creek Technical Specifications and the Corrective Action Program.

VII. Conclusion

The Action Request Process is in place at Hope Creek. The process is the focal point for many problem identification modes, and, based upon programmatic assessments and performance indicators, appears to be functioning effectively. While the process is always evolving through continuous self-assessment and enhancements, the process provides confidence that when problems are identified, timely action is taken. The extent and cause of the problem is determined and evaluations for operability and reportability impacts are included. This is a key process that allows us to identify and resolve the thousands of actions required to assure the safe, reliable and cost effective operation of our nuclear power plants.

Response to Question E

Provide an assessment of the overall effectiveness of the current processes and programs in concluding the plant configuration is consistent with the design bases.

The foregoing sections of this response provide descriptions of configuration control processes at the Hope Creek Generating Station, including the process for problem identification and corrective action implementation. These processes, along with the identified validation reviews and independent assessments which evaluate process effectiveness, pr⁻ ,de PSE&G with reasonable assurance that both plant procedures and configurations are in conformance with the Hope Creek design bases.

In evaluating the information provided in "Response to Questions A through D", PSE&G believes sufficient information has been provided to conclude with reasonable assurance that the Hope Creek Generating Station is configured and being operated, and maintained within its design bases. As additional issues are discovered through continuing review activities, they will be addressed and resolved in accordance with our corrective action program and plant Technical Specifications.

The above conclusion is primarily based upon the following information:

1. An Independent Design Verification Program (IDVP) provided assurances that the Hope Creek design had been adequately implemented prior to receiving the plant operating license. The IDVP was conducted by a large team of experience engineers with numerous oversight visits from NRC engineers and consultants. IDVPs were not performed on most power plants but only those licensed in the mid-1980 time frame. Consequently, the design bases documentation for Hope Creek was well above the average base-line of most power plants as of the mid-1980s.
2. The Hope Creek design was compared to the as-built configuration during the 1991 to 1994 time frame, when Configuration Baseline Documents (CBDs) were prepared for the 29 systems with primary safety significance. This activity provides reasonable assurance that design bases information is adequate.

3. Deficiencies identified coincide with CBD preparation were resolved (with some minor items still being resolved) in accordance with the corrective action program.
4. The configuration control process in effect since CBD preparation has been generally effective in maintaining design bases consistency, as demonstrated by recent assessments and inspections. In addition since the Hope Creek and Salem Generating Station control processes are common, the significant number of reviews and validations performed within the last two years at Salem provide additional confidence in the Hope Creek process.
5. The current processes used to maintain and control changes to the design have been upgraded and based upon recent assessments, have been shown to be adequate.
6. Various design bases assessments including the Electrical Distribution System Functional Inspection (EDSFI) and the in-progress Service Water System Operational Performance Inspection (SWSOPI) provide reasonable assurance that the consistency between the design bases and plant procedures and configuration has been maintained. [Note: The service water portion of the SWSOPI is complete; the Safety Auxiliaries Cooling System (SACS) portion is not complete.]
7. As additional issues are discovered through continuing review activities, they will be addressed and resolved in accordance with our corrective action program and plant Technical Specifications.
8. PSE&G has an effective corrective action program in place to identify and resolve design bases problems in a timely manner.

In general, PSE&G believes the evidence indicates our corrective action program, document control program, and configuration control process are strong, but that UFSAR consistency can continue to be improved and engineering staff design bases knowledge can be improved. Based on these observations, a program to continue design bases related improvements is outlined in Attachment 2 of this letter.

HOPE CREEK GENERATING STATION

Response to 10 CFR 50.54(f) Letter Dated October 9, 1996

ATTACHMENT 2

HOPE CREEK GENERATING STATION
Response to 10 CFR 50.54(f) Letter
Dated October 9, 1996

DISCUSSION OF FUTURE ACTIONS / COMMITMENTS

Attachment 1 provides the rationale for our reasonable assurance that design bases requirements are translated into the plant configuration and operating procedures for the Hope Creek plant. This section provides a brief discussion of both ongoing and future actions that will be completed over the next two years to provide additional confirmation of compliance with design bases. The detailed plans for conducting these further assessments, including specific information on scope and schedule, will be provided to the NRC within 60 days of the date of this letter.

<u>Ongoing Activities</u>

There are several ongoing initiatives discussed in this response that are in progress or will be completed (some of these may be rolled into the future reviews discussed below). These include the following:

- Configuration Baseline Document (CBD) Validation Program.
- Service Water System Operational Performance Inspection (SWSOPI). The service water portion of the SWSOPI is complete; the Safety Auxiliaries Cooling System (SACS) portion is not complete.

<u>New Initiatives</u>

In addition to the completed and ongoing activities to provide reasonable assurance of compliance with design bases, PSE&G will embark on a comprehensive Design Bases Review project. This project will focus on areas of review not yet covered in previous assessments and weaknesses discussed in this response. These reviews will be prioritized based on the safety significance of the plant systems using the Maintenance Rule definitions. Systems will be categorized according to their risk significance (e.g., Safety Analysis, Risk Significant, Risk Important Other systems). The initial focus will be on the risk significant and safety analysis systems. The scope of the project will be changed as appropriate based on findings (e.g., may shift the focus if appropriate). The desired end state of this project is expected to be a fully documented design / licensing bases which have been validated through plant procedures and is easily accessible for use in day to day conduct of work. To the extent that they have not already been accomplished by completed or ongoing activities items that will be evaluated as part of this project to ensure consistency with the design bases include:

- Reviews of UFSAR versus plant configuration and operations

- Key SER reviews for consistency with UFSARS
- Regulatory Commitment review
- Setpoint limits
- Verify testing adequacy
- Reviews of plant procedures against design bases documentation (e.g., UFSAR)
- Configuration walkdowns
- Safety System Functional Review type Vertical Slice reviews for selected systems
- Review of system lineups
- Drawing reviews

The approach planned for these reviews is to maximize the use of PSE&G resources to ensure clear ownership and improve the understanding of design bases by PSE&G personnel. Multidisciplinary teams will be formed to perform a comprehensive review of design bases information on a system / structure basis. Wherever possible, these teams will be structured to comprise the system manager, system design engineer, system senior reactor operator and other cross-disciplinary engineering support as may be required by the particular system (e.g. specialty engineering). The team will then (1) reconfirm or identify design bases requirements and related system attributes, (2) initiate changes to the CBD, as appropriate, including updating from the recent outage and (3) initiate UFSAR change requests (including the related safety evaluation) to clearly delineate both the design bases requirements and system attributes which implement the design bases.

In conclusion, PSE&G plans to undertake a comprehensive Design Bases Review project which will complement the work already completed to provide further confirmation that the Hope Creek plant is operated in accordance with it's design bases. The specific details relative to the scope and schedule of this project will be provided in a follow-up letter within 60 days from the date of this response.