Drew B. Fetters
Vice President
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PECO NUCLEAR
A UNIT OF PECO ENERGY

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NRC 10 CFR 50.54 (f)

February 4, 1997

Docket Nos. 50-277
50-278
50-352
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License Nos. DPR-44
DPR-56
NPF-39
NPF-85

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Peach Bottom Atomic Power Station, Units 2 and 3
Limerick Generating Station, Units 1 and 2
Response to NRC Letter from James M. Taylor (USNRC) to C. A. McNeill Jr. (PECO Energy) dated October 9, 1996.
"Request for Information Pursuant to 10 CFR 50.54 (f) Regarding Adequacy and Availability of Design Bases Information"

Gentlemen:

By letter dated October 9, 1996 from J. M. Taylor (USNRC) to C. A. McNeill Jr. (PECO Energy), "Request for Information Pursuant to 10 CFR 50.54 (f) Regarding Adequacy and Availability of Design Bases Information," the Nuclear Regulatory Commission (NRC) required that, within 120 days of the date of receipt of the letter, PECO Energy submit a written response addressing five specific information requests. In addition, the letter requests a description of any ongoing or planned design review or

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reconstitution programs. This letter is PECO Energy Company's response to the subject letter for Peach Bottom Atomic Power Station, Units 2 and 3, and Limerick Generating Station, Units 1 and 2.

PECO Energy appreciates the need to maintain configuration control of the design and design bases of Limerick Generating Station and Peach Bottom Atomic Power Station. Management's expectations are clearly stated in a Configuration Management Policy and a Configuration Management Directive. Our programs and processes have not been static, but rather have evolved in response to ongoing internal reviews as well as industry and NRC developments. By remaining current on industry and NRC developments, PECO Energy has been able to enhance its programs and processes to incorporate advances in existing guidance. There have been a significant number of configuration control and design bases verification programs implemented by PECO Energy in response to various industry events, industry initiatives, and corrective actions in response to problems identified by both internal and external assessments. Examples include the development of Design Baseline Documents (DBDs), the Updated Final Safety Analysis Report (UFSAR) Verification Project, the implementation of the Improved Technical Specifications (Peach Bottom Atomic Power Station), and the development of the Component Record List.

This response describes how PECO Energy maintains and adheres to the plants' design bases, and also discusses our programs for maintaining the adequacy and availability of design bases information. The term "design bases," as used in the request, is defined in the same manner as in 10 CFR 50.2: "Design bases mean 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 ...". The design bases of a facility, as so defined, is a subset of the licensing bases and is contained in the UFSAR. Information developed to implement the design bases is contained in other documents, some of which are docketed and some of which are retained by PECO Energy. The design bases for each facility forms the legal basis by which compliance with NRC requirements is to be judged.

PECO Energy is mindful of the broader concerns identified by the NRC with regard to accuracy of the UFSAR and adherence to the plants' licensing bases. The PECO Energy processes do not necessarily differentiate between the design and licensing bases for the control mechanisms. Therefore, in order to extract the most benefit and learning from our internal reviews performed in response to this request and to be fully responsive, the review performed by PECO Energy did not attempt to differentiate between the licensing bases, design bases, and other supporting design information.

PECO Energy has performed an extensive assessment in order to provide a complete and thorough response to this request for information. This assessment reviewed the current configuration management programs and controls, the translations of the design bases into the appropriate operating, maintenance and testing procedures, the performance of the Systems, Structures, and Components (SSCs) as it relates to the design bases, the problem identification and corrective action processes, the results of the extensive internal and external assessments of the configuration management program and processes, and the efforts which PECO Energy has pursued in response to various industry events and initiatives and NRC developments. This assessment was led by a core team which was responsible to plan, organize, provide oversight and facilitate the review performed by task teams assembled to review each aspect of the subject information request.

The task teams consisted of experienced engineers and professionals who reviewed the processes, procedures and the results of internal and external assessment activities (including inspections, surveillances, audits, Safety System Functional Inspections (SSFIs), and self-assessments) for the specific areas outlined in this request for information. As a result of this review, each task team provided written discussions and rationale for the conclusions specific to its assigned tasks. The task teams were specifically charged to review existing information available to support their evaluation of the assigned task. Each task team provided a multi-disciplined assessment of their specific task. The core team provided a challenging multi-discipline management review of team conclusions and rationale. An additional review was provided by a designated Review Panel comprised of a cross section of senior management from both sites, the Station Support Department, Nuclear Quality Assurance Department, and Legal Department, as well as two external consultants. Four members of the Review Panel are also members of the PECO Energy Nuclear Review Board. This Review Panel provided another challenging review of the task team methodology and of the information sources that provided support for statements of fact and conclusions. An open issues identification and resolution process was used to capture the Review Panel issues and track them to resolution. This assessment did identify areas where our processes could be improved. These improvement opportunities are being evaluated and appropriately pursued.

In order to prov. [illegible] a clear and logical response to this request, PECO Energy has provided detailed responses, in Attachment 1, to each of the five specific information requests and, in Attachment 2, to the additional question regarding design review or reconstitution programs. Attachments 1 and 2 discuss numerous improvement and corrective action items which were identified and evaluated during the preparation of this response, many of which are ongoing. In addition, the preparation of the response identified actions which PECO Energy believes should be implemented to further improve the programs and processes discussed in those attachments. To eliminate any ambiguity as to the commitments which PECO Energy is making as part of this response, Attachment 3 lists and describes each such commitment and the text of Attachments 1 and 2 contains a reference to Attachment 3 where appropriate.

Attachment 4 contains a Glossary of Terms and a list of Acronyms and Abbreviations used in this response. All four attachments refer to PECO Nuclear, which is a unit of PECO Energy.

The procedures, processes, and programs discussed in this response are dynamic by nature and, while the discussion of the specifics provides an accurate description of the present state of each, PECO Energy will continue to revise them in accordance with the approved revision processes and the applicable regulatory requirements, without modification of this response. Information (e.g., program documents, procedures) referred to in this response is contained in docketed correspondence or are available for your review.

The PECO Energy Configuration Management Directive establishes the Vice President of the Station Support Department as having responsibility for Configuration Management. I, as Vice President Station Support Department, have been designated to respond on behalf of PECO Energy. An appropriate affidavit is enclosed.

The results of this assessment conclude that the existing PECO Energy configuration management processes, including the corrective action process, provide reasonable assurance that the configuration control of the design of Limerick Generating Station and Peach Bottom Atomic Power Station is adequate and that the plants are maintained and operated in accordance with their design bases.

If you have any questions regarding this submittal, please contact me.

Very truly yours,

D. B. Fetters
Vice President, Station Support Department

cc: H. J. Miller, Administrator, Region I, USNRC
W. L. Schmidt, USNRC Senior Resident Inspector, PBAPS
N. S. Perry, USNRC Senior Resident Inspector, LGS

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA :
: ss
COUNTY OF CHESTER :

DREW B. FETTERS, being duly sworn according to law, deposes and says as follows:

1. I am Vice President, Station Support of PECO Energy Company which is authorized by operating licenses issued by the U.S Nuclear Regulatory Commission to operate Limerick Generating Station Units 1 and 2 and Peach Bottom Atomic Power Station Units 2 and 3.

2. I am authorized to sign PECO Energy Company's response to the U. S. Nuclear Regulatory Commission's letter dated October 9, 1996 requesting additional information pursuant to 10 CFR §50.54(f) regarding the adequacy and availability of design basis information which is contained in the preceding letter dated February 3, 1997 and the attachments thereto (the "Response").

3. The Response was prepared by a core team of individuals including experienced engineers and other professionals under my management. The core team divided the Commission's request for additional information into tasks which were assigned to multi-disciplined task teams consisting of experienced engineers and professionals who reviewed the applicable processes, procedures and other detailed information, and provided written discussions and rationale for the conclusions specific

to each assigned task. Each task team submitted their written reports and conclusions to the core team which provided a challenging multi-discipline management review of the teams' reports and conclusions. In addition, review was provided by a designated review panel comprised of a cross section of senior management from Limerick and Peach Bottom, the Station Support Department, Nuclear Quality Assurance Department, and Legal Department as well as two external consultants.

4. I have read the attached Response and in reliance on that review, my inquiries of the individuals involved in the preparation of the Response, and the processes and reviews discussed in the preceding paragraph, and independent oversight, do hereby affirm that the contents of the accompanying Response are true and correct to the best of my knowledge, information and belief.

DREW B. FETTERS

Subscribed and Sworn to
before me this 4th day
of February, 1997.

Mary Lou Skrocki
Notary Public

Notarial Seal
Mary Lou Skrocki, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires May 17, 1999
Member, Pennsylvania Association of Notaries

ATTACHMENT 1

REQUEST (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.

RESPONSE

The PECO Nuclear processes for engineering design and configuration control, including those that implement 10CFR50.59, 10CFR50.71(e), and applicable 10CFR50 Appendix B requirements, are described below. The combination of PECO Nuclear engineering design and configuration control processes are hereafter referred to as Configuration Management (CM).

I. CONFIGURATION MANAGEMENT OVERVIEW

CM is described in PECO Nuclear Policy, NP-CM-1, Rev.1, "Configuration Management" and PECO Nuclear Directive, ND-CM-1, Rev.1, "Configuration Management." This policy and directive provide the framework for management of design activities, plant configuration, and document configuration. CM is an integration of procedural, programmatic, and management controls designed to assure that:

- The Limerick and Peach Bottom plants conform to the approved design and licensing requirements.
- The physical and functional characteristics of the plants are accurately reflected in controlled documents.
- The status of design, plant and functional design changes, temporary plant alterations and associated documents are readily accessible to appropriate line organizations.

CM is embedded within the requirements, commitments, and good practices of PECO Nuclear functional area work processes. CM is implemented and maintained via functional area procedures and processes. CM feedback and evaluation mechanisms such as assessments and performance indicators are established to monitor and improve performance.

The following are key to effective CM within PECO Nuclear:

- The Station Support Department is the single Design Authority (DA) responsible for technical excellence of the design process and establishment and maintenance of design requirements and controls.
- A common integrated information management system for CM processes, the Plant Information Management System (PIMS), is utilized.
- Common CM processes and procedures are utilized except for specialized CM activities.
- CM problem identification and resolution mechanisms are clearly established.
- Oversight mechanisms such as assessments and performance indicators that monitor design and configuration control activities are in place.

- Training programs to assure personnel are cognizant of design and configuration control elements in their respective functional areas are established.

II. DETAILED DESCRIPTIONS OF ENGINEERING DESIGN AND CONFIGURATION CONTROL PROCESSES

This section describes the PECO Nuclear process for engineering design control and the PECO Nuclear process for configuration control. Included in these descriptions are the elements of the processes that implement 10CFR50.59 and 10CFR 50.71(e) and applicable requirements of 10CFR50 Appendix B. These descriptions are summaries of our design and configuration control processes provided to convey a sense of how they work and are not intended to provide all procedural details. Also described are the problem reporting mechanisms used in the engineering design and configuration control processes, CM related training and CM oversight.

The design and configuration control philosophy which is established by PECO Nuclear policies and directives is carried out by implementing procedures. For the purposes of this report, the term procedure is used for procedures, guidelines, or manuals.

A. Engineering Design Control Methods

This section presents a discussion of PECO Nuclear's processes that make changes to the design or may impact the design bases. The philosophy for making changes is established by management through a set of policies and directives and performed by the workforce in accordance with implementing procedures. The discussion identifies the PECO Nuclear functional areas that make design changes, the design bases change processes, and the governing procedures.

The management expectations for processes that change the plant design bases are delineated in a set of controlled policies and directives. These policies and directives govern activities in the Design Bases Maintenance, Fuel, Licensing, Plant Change Process, Nuclear Information Systems, Emergency Preparedness, Radiological Environmental and Meteorological (REM) Monitoring, Chemistry, Security, and Facilities Management functional areas. These ten functional areas perform activities that may affect the design bases by making changes to any of the following: plant design or Facility Operating License (FOL), computer programs, Emergency Plan, Offsite Dose Calculation Manual (ODCM), fuel design, or Security Plan. A key element to maintaining control in these areas is the designation by policy NP-MD-2 of the Station Support Department as the single Design Authority (DA). The DA is responsible for the technical excellence of the design process and establishment and maintenance of design requirements. Therefore, the engineering and design process used by the PECO Nuclear organizations is under the control of the Station Support Department. The change processes for each area mentioned above are discussed in the following sections.

1. Plant Design and Facility Operating License Change Process

PECO Nuclear's commitments for design control are contained in the PBAPS and LGS QA Program Descriptions. Both plants commit to Regulatory Guide 1.64, which endorses ANSI N45.2.11-1974.

All activities are performed in accordance with governing written procedures. The governing procedures invoke implementing procedures providing the detail to assure that activities are comprehensively performed. To the extent practical procedures are common.

ATTACHMENT 1

In order to enhance our work processes, PECO Nuclear implemented a common integrated computer based process called the Plant Information Management System (PIMS). PIMS terminals are located at all PECO Nuclear facilities, providing controlled access to centralized data for all organizations.

The engineering design functions are performed primarily in the Document Control, Management Action, Resource Data, and Inventory Control PIMS modules. Design change packages are developed and published as Engineering Change Requests (ECR) under the Document Control module. The text associated with the design change is contained in the computerized record while drawing and document revisions are hard copy attachments to the ECR. These hard copy attachments and the computer printouts are maintained as nuclear records. The ECR also controls changes to information in the Resource Data module and the Document Control Register by automatically updating them after the design change is installed. The Resource Data module contains the plant Component Record List (CRL), which defines critical component data such as component number, safety class and component qualification classifications. In addition, to facilitate the work control process, the CRL contains other information such as associated drawings, spare parts information, and procurement information. The Document Control Register contains record information for all controlled drawings and documents such as title, revision number, identification of pending design changes, as-building category, and drawing/document owner. The Inventory Control module contains the Inventory Parts Catalogue (IPC), specifications, Bill of Material (BOM), and inventory levels for equipment and materials.

The nature of the design change being made is indicated by ECR type. A listing of the different forms of design changes, the type of ECR used, and the applicable governing procedure is shown below.

Change Type	ECR Type	Procedure
• Modifications	MOD	MOD-C-3
• Modification Design Revisions	MDCH	MOD-C-3
• Minor Physical Changes	MPC	MOD-C-3
• Temporary Plant Alterations	TPA	MOD-C-7
• Setpoint Changes	ISCR	MOD-C-8
• Design Equivalent Changes	DEC	NE-C-270
• Nonconformances	NCR	A-C-901
• Document Change	DCR	NE-C-440
• Material Evaluations	MEVL	NE-C-270

The design change package is published as an ECR in accordance with procedure MOD-C-9, Engineering Change Requests. This procedure requires that a 10CFR50.59 Review be performed for applicable design changes. When using drawings/documents, personnel are required by procedure to verify that they are using the latest controlled revision as identified in the Document Control Register. The combination of a governing procedure, the ECR procedure, and implementing procedures establishes a comprehensive process to identify, analyze, and revise the design bases. Following is a list of the major activities and controlling procedures of the design change process that assure maintenance of the design basis:

• Acceptance Testing	MOD-C-5
• ALARA	HP-C-301
• As-building Drawings/Documents	NE-C-440
• ASME Code Impacts	A-C-80
• Cable & Raceway Management	NE-C-320
• Calculations	NE-C-420
• Commitment Tracking	LR-C-1

ATTACHMENT 1

• Component Classification	NE-C-210
• Component Record List (CRL)	NE-C-211
• Configuration Control of Digital Systems	A-C-135
• Design Baseline Document (DBD) Change	NE-C-230
• Design Input Document	NE-C-205
• Design & Drafting Standards	NE-CG-400
• Digital Upgrade Evaluation	NE-CG-936
• Dynamic Qualification	NE-C-220
• Electrical Load Changes	NE-C-310
• EMI Evaluation	NE-CG-926
• Environmental Qualification	NE-C-220
• Fire Protection	NE-C-250
• Failure Modes & Effects Analysis	NE-CG-270
• Hazard Barriers	NE-CG-265
• Independent Review	MOD-C-9
• Nuclear Plant Reliability Data System (NPRDS)	AG-CG-14
• Station Procedures & Programs Impact	MOD-C-5
• Software Development	IM-C-2
• Software Verification and Validation	IM-CG-4
• Specifications	NE-C-110
• Technical Specification (TS) or Bases Change	LR-C-8
• Training Bulletin to Operations	MOD-C-5
• UFSAR Change	LR-C-9
• Walkdowns	MOD-CM-1
• 10CFR50.59 Review	LR-C-13

In addition to the guidance provided within the procedures themselves, guidance on performing modifications is provided in a Modifications Manual, MOD-CM-1. The manual provides guidance for modification activities covering such topics as lead personnel responsibilities; development of conceptual design; ANI inspections; acceptance test criteria and acceptance test development; guidance for conducting walkdowns; and, identification and revision of affected station documents, programs, and procedures.

Changes to the Facility Operating License (FOL), Technical Specifications (TS), and TS bases are controlled through procedure LR-C-8 which establishes the requirements and responsibilities for initiating and processing changes to these documents and is applicable to all proposed and issued changes to the nuclear facilities. Upon identification of a proposed change to the TS, an ECR which identifies and describes the affected TS and/or TS bases is initiated in accordance with MOD-C-9. LR-C-8 ensures that the Licensing Change Request (LCR), i.e., the ECR with the applicable marked-up sections, is reviewed and approved by the appropriate PECO Nuclear organizations including the Nuclear Review Board (NRB) before being submitted to the NRC for approval as a License Change Application (LCA). The procedure requires that a 10CFR50.59 Review and a No Significant Hazards Consideration (NSHC) per 10CFR 50.92 be prepared for the change. Procedure NRB-10 gives guidance to the NRB in review, approval, and disposition of FOL, TS, and TS bases changes.

Changes to the UFSAR are controlled by procedure LR-C-9 which establishes the requirements and responsibilities for the initiation, processing, and distribution of changes to the UFSAR. This procedure is applicable to all proposed and approved changes to the nuclear facilities. These changes may involve changes to procedures, structures, systems, or components. An UFSAR

ATTACHMENT 1

change is processed using the ECR process described in MOD-C-9 and includes the approved 10CFR50.59 Review for the activity and marked-up pages of the UFSAR. Procedure LR-C-9 reflects the requirements of 10CFR 50.71(e). The UFSAR is revised and updated consistent with the requirements of 10CFR 50.71(e). Procedure LR-UG-2 provides guidance to the Licensing group in processing of UFSAR changes.

Procedure LR-C-13, "10CFR50.59 Reviews", establishes the requirements for determining if a facility change, test, or experiment constitutes an Unreviewed Safety Question (USQ) or requires a change to the FOL or TS. The procedure applies to activities that may make a change to the facility or procedure as described in the SAR (see Attachment 4), may violate a commitment stated in the SAR, or may conduct a test or experiment not described in the SAR. LR-C-13 implements the requirements of 10CFR50.59 and is supported by procedure LR-CG-13 which provides guidance on how to perform the review and defines the limited circumstances when a 10CFR50.59 Review is not required. This exclusion criteria is limited to the following changes:

- Editorial or typographical corrections where the change does not affect the scope, results, or requirements of the document.
- The change is intended to resolve conflicts between the SAR and actual plant design where the SAR is correct and the plant is incorrect
- Incorporation of certain changes such as general location change or label changes from an approved ECR.
- Changes which solely impact the QA Program Description, the Emergency Plan, or the Security Plans which are subject to 10CFR50.54.
- Minor changes to SAR figures such as a redraw of an existing drawing or incorporating changes evaluated by previous 10CFR50.59 Reviews.

The 10CFR50.59 Review process consists of two distinct stages. The first stage is a 10CFR50.59 Determination which evaluates whether the activity makes changes to the Technical Specifications, the FOL, procedures as described in the SAR, or the facility as described in the SAR, or involves a test or experiment not described in the SAR. The Determination requires answering four specific questions addressing those areas. If it is determined that no change to the Technical Specifications, the FOL, or the SAR is required, and the activity does not involve a test or experiment not described in the SAR then the Determination forms the basis for concluding that the activity does not create a USQ and the Determination constitutes the 10CFR50.59 Review. If it is determined that any of the four questions is answered in the affirmative, then a Safety Evaluation (second stage) is required.

The 10CFR50.59 Safety Evaluation determines whether the activity involves an USQ by answering the questions contained in 10CFR50.59 (a)(2). This evaluation is required to address seven specific questions to determine:

- if the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

- if the possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

- if the margin of safety as defined in the basis for any technical specification is reduced

If the Safety Evaluation concludes that there is no impact, then the Safety Evaluation forms the basis for concluding that the activity does not create an USQ. If there is an impact, then the activity must be handled in accordance with the requirements for an USQ. 10CFR50.59 Reviews

ATTACHMENT 1

receive a peer review to ensure completeness and accuracy.

The procedure for Temporary Plant Alterations (TPA) includes the following unique steps to address their special needs :

- Identification of plant conditions for which design requirements would not be met (so that those conditions are not entered)
- Prescription of an operational verification method to be performed following TPA installation and subsequent to removal to assure proper component/system operation
- Tagging of equipment & control switches with TPA information
- Marking-up of drawings in main control room, when required
- Conducting operator briefings
- Identifying compensatory actions as appropriate
- Revising the ECR to authorize removal of the TPA, document change postings, and marked-up drawings in the main control room
- Performing quarterly walkdowns for TPAs installed greater than 90 days to verify that installation & tagging remain intact

Design change packages developed using the plant design change process are comprehensive in their treatment of the design bases. However, recognizing that this is a difficult task for complex changes, a supplemental review that directly supports the goals of assuring that the change has been correctly designed, completely tested and conforms to the evaluated configuration was added to the process in 1995. The supplemental review was developed as a corrective action to improve the overall quality of modification packages. In this process, appropriate design changes go through a screening step to identify those having a level of complexity warranting the supplemental review after design, installation planning, and acceptance test development are completed. The process includes a review to assure that all components affected by the design change are tested and that design goals are met.

2. Specialized Change Processes

Some specialized change processes for areas with unique requirements use evaluation methods in addition to or different than the ECR process described above, but all change processes incorporate consideration of the effect of a change on the design bases.

a) Computer Program Design Change Process

Computer program development and changes are controlled by procedures which require that a 10CFR50.59 Review be performed in accordance with LR-C-13 and LR-CG-13. Through the use of procedure IM-C-2, software is classified according to its plant safety function, operational requirements, or business needs. Application software is classified according to the following criteria:

- Class 1: operates in a plant system that is safety related and provides direct output control functions.
- Class 2a: operates in a non-safety related plant system and provides direct output control functions.
- Class 2b: includes application software which performs process monitoring, performs engineering calculations, is used in reactor physics, implements a licensing commitment, satisfies a TS requirement, or is used for radiochemistry analysis.
- Class 3: application software which does not meet the criteria for Class 1, 2a, or 2b.

ATTACHMENT 1

The classification determines the level of control used in the development and maintenance of the software.

Vendor developed programs for digital based plant instrumentation and process systems are classified as Class 1 or 2 and are handled in accordance with A-C-135 and NE-CG-936. Using procedure A-C-135, the various kinds of programs (operating systems, application programs, etc.) are assigned a document number and controlled in the Document Control Register similar to a drawing. Firmware (i.e., PROM based) programs are controlled like hardware by identifying their version number in the IPC for the equipment that contains the firmware. Both software and firmware are approved and controlled using the ECR process. Procedure NE-CG-936 provides the guidance to assure that digital upgrade modifications address the appropriate recent industry standards. Physical design changes made to plant computer hardware which is part of a plant modification are implemented in accordance with the ECR process.

Other software is handled in accordance with IM-CG-4 and IM-C-5. Procedure IM-CG-4 requires that, for Class 1 or 2 software, a Verification & Validation traveler be used to assure procedural compliance; that a document update form be used to identify all affected documents; and that an independent review be performed. IM-C-5 provides the administrative controls for computer software error management. For Class 3 software, these steps are recommended but not required.

b) Nuclear Emergency Plan and Facility Design Change Process

Changes to the Nuclear Emergency Plan are controlled by procedure EP-C-1, Development & Maintenance of the Nuclear Emergency Plan and Emergency Response Procedures. This procedure requires that all changes be reviewed in accordance with 10CFR50.54(q) to ensure that they do not decrease the effectiveness of the plan. A Performance Enhancement Program (PEP) (see section I.C.3 of this response) issue has been initiated to evaluate an opportunity for improvement in the area of Emergency Preparedness design/document control process for the Emergency Response Facilities outside of the plant Protected Area Boundary. However, it has been concluded that all applicable Emergency Preparedness requirements are being met by the existing controls and processes.

c) Offsite Dose Calculation Manual (ODCM) Change Process

The requirements for changing the ODCM are contained in TS section 5.5.1.c for PBAPS and in ODCM Appendix C for LGS. LR-C-13 identifies the ODCM to be part of the SAR; therefore, changes are required to be evaluated by a 10CFR50.59 Review in accordance with procedure LR-C-13. That procedure also contains the controls necessary to ensure that the ODCM is updated with the Annual Effluent Release Report.

d) Fuel Design Change Process

Changes to the fuel design are controlled by approved procedures which provide assurance that fuel design changes are properly evaluated and controlled by requiring that:

- a 10CFR50.59 Review be performed
- the reference core design is used for reload licensing calculations
- reload licensing activities are performed and reported in the Supplemental Reload Licensing Report
- licensing results are reported in the Core Operating Limits Report
- thermal limit values are loaded into the Plant Monitoring System (PMS)

- the core loading pattern is documented in the Core Design Report
- detailed instructions for core management activities are transmitted to the site in the Cycle Management Report
- Core Component Transfer Authorization Sheets are based on the Core Design Report
- the as-loaded core is identical to the design specified in the Core Design report

e) Security Plan Change Process

Changes to the Security Plan are controlled by procedure SEC-C-6. This procedure requires that changes be evaluated under 10CFR 50.54(p) to ensure that the effectiveness of the Security Plan is not decreased. Changes to security structures are processed as plant design changes (ECR process).

B. Configuration Control Methods

This section describes the methods used for CM activities that contain, control, or utilize design bases information including how these methods satisfy the requirements of 10CFR50.59, 10CFR 50.71(e), and applicable criteria of 10CFR50 Appendix B. This section also covers the aspects of implementation and verification of the plant change control process. Other configuration control items such as operational plant status control are also discussed as they relate to the specific functional areas.

Procedures utilized in the functional areas discussed in this section are prepared utilizing several common procedures. These procedures are A-C-1, A-C-4, A-C-4.2, AA-C-5, and AG-CG-91. Procedure AA-C-5 establishes the requirements for preparation and revision of procedures. AA-C-5 requires the preparer of a new procedure or a procedure revision to ensure that the scope and content of the procedure continues to adequately implement requirements of the source documents including the SAR, non-SAR commitments, and PECO Nuclear policies and directives. For new procedures, AA-C-5 also requires preparation of a 10CFR50.59 Review if required in accordance with LR-C-13 and LR-CG-13. Procedure revisions receive a 10CFR50.59 Review if required by their revision control block which is defined during initial issuance of the procedure.

A-C-1 provides guidance on procedure content and format. AG-CG-91 provides guidance on word processing conventions to ensure adequate on-line display of procedural documents using the Text Management System.

A-C-4.2 establishes the Station Qualified Reviewer (SQR) and Quality Reviewer (QR) Programs. A-C-4.2 requires the SQR to review the 10CFR50.59 Review for the procedure revision. Both the SQR and QR are required to ensure that the revised procedure adequately implements requirements of the source documents, including the SAR, non-SAR commitments, and PECO Nuclear policies and directives.

AA-C-4 provides requirements for the format, content, and process for policies and directives.

1. Drawing/Document Control

The role of the Drawing/Document Control functional area is to distribute, control, and as-build the documentation changes so that users accessing the information are provided with the latest approved version. This is accomplished by procedures A-C-2, A-C-92, NE-C-440 and DC-C-3. A key to maintaining control over these areas is the use of the Document Control module in PIMS.

ATTACHMENT 1

The activities performed in this functional area are the result of documentation changes approved by the Plant Change Process and Design Bases Maintenance functional area. The following controls are in place which ensure drawing and document control:

<u>A-C-2 Distribution and Control of Procedural Documents</u>

- Procedure information updated & controlled by Document Control Register in PIMS
- Procedures available by networked computer (Lotus Notes)
- Verification of latest revision number via the Document Control Register prior to use of document
- Controlled print self-assessment audits
- Record retention in Nuclear Records Management System (NRMS)

<u>A-C-92 Vendor Documentation</u>

- Documentation approved and maintained via the ECR process
- Documentation updated and controlled by the Document Control Register in PIMS
- Document listed in the CRL record for associated components
- Document status defines whether document is under review or approved for use
- Vendor solicitation program to ensure continued accuracy of document
- Controlled print self-assessment audits

<u>NE-C-440 As-Building and Engineering Document Classification</u>

- Design changes authorized by ECR
- Documents are assigned as-building categories commensurate with importance and use
- Timeliness of as-building after work completed is determined by document as-building category
- "Information only" documents categorized to require verification before use
- As-building process can be automatically triggered by ECR work completion
- Partial as-building for work completed in stages

<u>DC-C-3 Drawing/Change Document Processing and Use</u>

- Document transmittal acknowledgment by receipt
- Controlled files stamped & maintained by single organization (Document Services)
- Safeguard documents specially controlled
- Station master drawing file maintained
- Superseded and voided drawings removed from controlled locations
- DCR distribution module assures file maintenance

2. Licensing Configuration Control

The Licensing functional area includes monitoring and evaluating information from the nuclear industry including such organizations as NRC, INPO and NSSS suppliers. Procedure LR-C-4 establishes an Operating Experience Assessment Program (OEAP) that ensures that this information is monitored on a regular basis and is disseminated to the appropriate PECO Nuclear organizations to determine applicability and required actions. Information monitored by the OEAP program may result in changes to the design/licensing bases being initiated through the MOD and/or LR series of procedures. Information which is published in the Federal Register is also monitored and disseminated through the use of procedure LR-UG-1.

ATTACHMENT 1

Changes made to the design/licensing bases may necessitate the revision of certain plant procedures and/or programs. Procedure LR-CG-6 provides the measures to control the identification, implementation, and documentation of changes to plant procedures and programs, and training as a result of revisions to license documents and other information formerly contained in the Technical Specifications. Approved changes to the UFSAR are processed for inclusion in the 10CFR50.71(e) UFSAR update in accordance with LR-C-9.

Design and licensing bases changes identified by the OEAP or other programs may result in commitments being made by PECO Nuclear to the NRC or other governmental/industry authority. Procedure LR-C-1 describes PECO Nuclear's Commitment Tracking Program (CTP), and establishes the responsibilities, authorities, process, and organizational interfaces for tracking and assuring compliance with these commitments. The CTP has its own primary module in PIMS.

3. Verification of Configuration Changes

During design of modifications, required inspections, such as those required by ASME, may be identified. During the planning of Work Orders, planners establish the appropriate verifications in accordance with A-C-33 and the modification package. A-C-33 provides direction for establishing worker, independent, supervisory, quality, and double verifications. NQA-4 establishes the requirements for implementation of NQA's Quality Verification Program including the establishment of NQA Quality Verification points.

4. Emergency Preparedness Configuration Control

Control of the design bases in the Emergency Preparedness functional area is accomplished using 10CFR50.54(q). Procedures are in place which require all changes to the Nuclear Emergency Plan to be evaluated to ensure that they do not decrease the effectiveness of the plan (EP-C-1, EP-UG-1). This demonstrates that the requirements of 10CFR50 Appendix E "Emergency Planning and Preparedness" remain satisfied. Documents which identify offsite structures, systems and/or components are maintained by the Emergency Preparedness organization.

5. Security Configuration Control

In the Security functional area, configuration control of Safeguards Information is addressed by procedure SEC-C-4. This procedure defines the administrative controls and requirements for the preparation, receipt, identification, use, reproduction and storage of Safeguards Information, some of which is design/licensing bases information. This procedure addresses the requirements of 10CFR73.21. Documents including drawings, vendor manuals and specifications which describe Security structures, systems and/or components are controlled by the ECR process.

Recently, weaknesses have been identified in the area of access control of Safeguards Information at PECO Nuclear. Specifically, certain Safeguards Information was found to be accessible to personnel who were not authorized individuals. A formal investigation of this issue was initiated through the PEP process and corrective actions have been initiated. In addition, PECO Nuclear has responded to a NRC "Apparent Violation" letter which was issued as a result of several NRC inspections performed at LGS, PBAPS, and PECO Nuclear headquarters (Chesterbrook).

ATTACHMENT 1

6. Information Management Configuration Control

Configuration control of vendor developed programs for digital based plant instrumentation and process systems are governed by procedure A-C-135 as described in Section II.A.2.a).

Configuration control for other software in the Information Management area is maintained by several different procedures. Procedure IM-C-8 establishes the process for the creation, collection, storage, and maintenance of Electronic Quality Records, e.g., PIMS Work Orders. Control of data and/or parameters that reside in files or databases and directly support the operation of the software is governed by IM-CG-14. This procedure establishes the components of data control and is applicable to data in Class 2b software systems. Procedure IM-CG-7 identifies the process for initiating requests to acquire, enhance, develop or modify PECO Nuclear software and/or hardware resources. The Information Service Request (ISR) is the mechanism used to accomplish these activities and is implemented through the PIMS Management Action module. ISRs which result in software design and licensing bases changes are used in conjunction with IM-C-2, IM-CG-4, and IM-CG-14 in addition to the MOD and LR series of procedures to assure that the design/licensing bases are maintained.

A prior review of Performance Enhancement Program (PEP) issues and assessments revealed that the area of software quality needs improvement. Specifically, problems were encountered in certain changes to PIMS software due to incomplete design requirements and inadequate testing of software interfaces. In the plant process computer area, some problems have been related to incorrectly configured database elements which resulted in NSSS software modules halting. In both of these areas, PEP issues were generated to identify root causes and corrective actions have been initiated. In mid-1996, there was a management initiative to improve software quality by performing an in-depth review of current software management practices. In support of this initiative, a review was recently completed by an independent consultant which concluded that improvements are needed in the area of software design, interfaces, testing and data control. Recommendations were made to strengthen software quality assurance procedures and to adopt an industry accepted software development model in order to create a more rigorous software CM process. The assessment of the recognized weakness of software configuration control will be completed and appropriate corrective actions implemented. Corrective actions associated with the software configuration control concerns will be identified and tracked via the PEP process (see commitment 5 in Attachment 3). In addition, NQA and ISEG personnel have been actively involved in several assessments to ensure root causes are determined and corrective actions identified.

7. Fuel Management Configuration Control

Core management activities, which utilize or control design bases information, are coordinated with Fuel Management processes per FM-C-1 and FM-400. Procedure FM-201 requires that analysis of the critical characteristics, i.e., mechanical, neutronic, and thermohydraulic, of each new fuel design be performed in conjunction with all potentially affected work groups prior to the implementation of the design.

FM-102 specifies the required content of Fuel Management design record files. FM-102 also describes how to index and organize the files and prepare them for plant life retention.

A specialized Fuel Management document control process has been established to meet the unique requirements of the Fuel Design change process described previously. Recently, opportunities for improvement were evaluated under the PEP process and corrective actions to improve Fuel Management document control are in progress.

ATTACHMENT 1

8. Health Physics Configuration Control

In the Health Physics functional area, procedures incorporate requirements that are contained in the UFSAR, Technical Specifications, ODCM, Reg. Guides, commitments, etc. A-C-100 is the governing procedure for the Radiation Protection Program and invokes 10CFR Parts 19, 20, 30, 50, and 71. The HP, HP-C, HP-CG, and HP-UG series procedures implement the current requirements.

The shielding program is governed by HP-C-313 which defines temporary shielding as shielding to be installed for less than six months. All other shielding is considered permanent and its installation is required to be evaluated using the ECR process. Shielding requests are documented using a Shielding Request Form (SRF). SRFs also serve as the tracking mechanism for installation and removal of shielding and are reviewed monthly by the site Shielding Coordinator.

Specification NE-048 establishes preapproved, limited case applications for temporary shielding. A 10CFR50.59 Review has been processed for the limited case applications allowed by NE-048. If a temporary shielding request meets the NE-048 criteria, it may be installed without additional Engineering review. If NE-048 criteria are not met, but Engineering has previously evaluated (including a 10CFR50.59 Review) the specific application, the temporary shielding may be installed without requiring additional Engineering approval. If a temporary shielding request does not meet the NE-048 criteria, nor has the shielding installation been previously evaluated, then the Shielding Coordinator requests Engineering to evaluate the new application. Engineering processes a 10CFR50.59 Review for the new application and establishes the specific requirements for installation of the shielding. Temporary shielding is either removed before it has been in place for six months, or an ECR is processed to evaluate it as a permanent installation.

9. Procurement Configuration Control

Procedures NE-C-270 and NE-CG-270 establish the engineering requirements for determining and documenting the technical and quality requirements of a plant item to be purchased. The Inventory Parts Catalog (IPC) is the repository of the technical and quality requirements to be invoked when purchasing a plant item. The IPC consists of Stock Code Numbers (SCNs) and each approved SCN provides the QA Class, the corresponding Purchase Class, Environmental Qualification Class, ordering information, the technical and quality requirements, storage requirements, shelf life, hazardous material requirements, and whether the item must be purchased from an approved vendor. 10CFR50.59 Reviews are performed for Design Equivalent Changes (DECs), for new plant items, and when reclassifying a piece part's QA or Environmental Qualification class. Bills of Material (BOM) are developed to link SCNs to plant components.

10. Operations Configuration Control

Operations personnel perform activities that directly change the configuration of the plant. These activities are controlled by following established procedures. The plant configurations established are within the limits specified in the UFSAR and the Technical Specifications. When plant conditions are established to conduct maintenance or troubleshooting, there are adequate controls and barriers in place within the procedures, i.e., Clearance and Tagging Manual, Operations Manual, Temporary Plant Alterations and Checkoff Lists, designed to prevent improper configurations from being established. Shift Management is required to continually evaluate plant status against Technical Specifications to assure that the design bases are not

ATTACHMENT 1

11. Maintenance Configuration Control

Procedure A-C-26 establishes controls for plant maintenance/modification work, corrective and preventive maintenance, Fix-It-Now and special processes (Welding, Chemical Cleaning, NDE). Work planning activities are processed using a PIMS Action Request and Work Orders. The Fix-It-Now teams are work teams that perform limited scope work on plant equipment. The planning process includes assuring that the material to be used is evaluated and qualified for the application by reviewing the CRL, BOM and IPC. The planning process also identifies the necessary implementing procedures, actions, testing requirements and verifications to assure configuration is maintained.

12. Testing Configuration Control

Plant testing procedures, such as post maintenance, modification, surveillance, routine and Plant Evolution Special Tests are performed to verify component and system operability. After satisfactory testing, the appropriate configuration is reestablished.

The Troubleshooting Program makes changes in system configurations to perform troubleshooting activities on plant equipment in accordance with A-41.1 (LGS) and A-42.1 (PBAPS). The Work Group Supervisor indicates on a Troubleshooting Control Form (TCF) whether a 10CFR50.59 determination is needed. If the TCF requires a Safety Evaluation, then that Safety Evaluation is presented to the Plant Operations Review Committee (PORC) per A-C-4 before performing the TCF. TCFs that require Safety Evaluations are evaluated to determine if a TCF is the appropriate administrative control mechanism to be utilized.

13. Chemistry and Radiological, Environmental and Meteorological Monitoring (REM) Configuration Control

Chemistry and REM procedures are developed to support appropriate operational configuration. Certain Chemistry procedures support compliance to the Technical Specifications.

The Offsite Dose Calculation Manual (ODCM) provides methods for calculating potential dose to members of the public outside the site boundary. The ODCM is revised to reflect changes to the radiological environmental monitoring program and to reflect changes to effluent monitoring equipment.

14. Radwaste Configuration Control

The Radwaste Process Control Program describes the interfaces, responsibilities, and requirements necessary to assure that waste generation, processing, packaging, storage, shipping and disposal activities are conducted in compliance with the facility design bases. Additional Radwaste procedures are developed to assure compliance to radwaste and hazardous material requirements specified in 10CFR, 29CFR, 40CFR and 49CFR.

15. Fire Protection Configuration Control

Procedure A-C-920 describes the requirements for the Fire Protection Program (Ref. LGS UFSAR Appendix 9A, Fire Protection Evaluation Report and PBAPS Fire Protection Program document) and assures compliance with license conditions and the fire protection commitments in the UFSAR. Changes to the Plant Fire Protection Program invoke the ECR change process and require a 10CFR50.59 Review in accordance with LR-C-13.

ATTACHMENT 1

C. Problem Reporting Methods

When problems are identified they are recorded, analyzed, and corrected through processes commensurate with the significance of the problem. The following provides a description of the problem reporting processes.

1. Equipment Trouble Tag (ETT)

Procedure AG-CG-26.1 establishes the instructions for identifying and documenting plant equipment problems using an ETT to physically identify the plant equipment problem and a Corrective Maintenance-ETT PIMS Action Request to document the problem resolution. Plant equipment problems that cannot be resolved by restoration to original configuration are processed in accordance with the Engineering Change Request Process (ECR) described below.

2. Engineering Change Request (ECR) Process

Discrepancies encountered during the maintenance process and problems discovered while using the ECR process are corrected using the ECR process. They are handled in one of two ways depending on whether or not the condition represents a nonconformance. Conditions that deviate from the design bases are processed as nonconformances and dispositioned in accordance with procedure A-C-901 as NCR-type ECRs. Conditions that are not nonconformances (e.g., problems discovered during installation or acceptance testing prior to placing the equipment in service) are dispositioned in accordance with procedure MOD-C-3 by either revising the ECR or initiating a new ECR. In all cases, the common elements of the ECR process are invoked to assure that corrective actions are proper and complete. For nonconformances, governing procedure A-C-901 adds steps to address those aspects that are unique to nonconformances.

Equipment problems identified during the procurement and receipt inspection activities are also processed under the ECR process. Procedure P-C-1 establishes the process for vendors to report problems to PECO Nuclear and to request approval of deviations from the technical or quality requirements of the purchase documents. If the technical or quality requirements require revision, either the original ECR is revised or a new ECR is processed. Procedure P-C-3 establishes the instructions for reporting equipment problems identified during the receipt inspection process. If the technical or quality requirements established for the purchased item require revision, either the original ECR is revised or a new ECR is processed in accordance with NE-CG-270.

The combination of the NCR and ECR procedures are designed to assure that nonconformances are properly evaluated, reported, and corrected and that the design bases configuration is maintained.

The NCR process is discussed in further detail in response to request (d).

3. Performance Enhancement Program (PEP) Process

Procedure LR-C-10, "Performance Enhancement Program (PEP)", defines a Condition Adverse to Quality (CAQ) as "A condition where procedures, work processes, or activities permit the potential for, contribute to, or result in failures, malfunctions, deficiencies, deviations, defective material or equipment, or noncompliance with specified requirements." Problems which create a CAQ are analyzed using the PEP process.

Corrective actions requiring design changes are accomplished using the applicable procedures.

The PEP process is discussed in further detail in response to request (d).

ATTACHMENT 1

4. Information Management (IM) Problem Reporting Methods

IM-C-5 provides the administrative controls for computer software error management and provides the steps to be followed to report, analyze, and document software errors. If a software error is identified, a PIMS Information Service Request (ISR) is initiated and routed to the Software Librarian who coordinates the resolution of the problem. If the problem resolution requires a software change, then the change is implemented using the process described previously. In addition, if software or firmware is controlled using procedure A-C-135, then the ECR process is used in the same manner as for correcting hardware type problems. As in all areas, if the problem creates a CAQ, then a PEP issue is also initiated.

Problems with certain plant computer hardware, such as data acquisition equipment, are handled in the same manner as problems with plant equipment, using the ETT, NCR, ECR, and PEP process as appropriate.

5. Emergency Preparedness (EP) Problem Reporting Methods

EP problems that are below the PEP threshold, but for which improvement is desired, are analyzed in accordance with an EP Action Item Tracking System defined by procedure EP-C-2. Items handled by the tracking system are below the NRC reportability threshold. Corrective actions are implemented using the Emergency Plan and Facility design change process described previously.

6. Fuel Management (FM) Problem Reporting Methods

Unique problem identification and reporting processes exist in procedures FM-200, FM-300, and FM-400 to assure that anomalies observed in past design work are considered as current fuel analyses are being performed. This allows past lessons learned to be brought forward in order to improve current and future designs. The data is controlled in Core Management design record files per procedure FM-102 and is disseminated during internal Fuel Management design reviews and at the Reactor Engineering Interface meetings. This data is also being used to develop core management performance indicators for presentation at Reactivity Management meetings which are held to ensure that reactivity stakeholders remain abreast of operational and engineering issues which can affect the ability to control core reactivity.

D. Configuration Management (CM) Training

The PECO Nuclear training process supports configuration management by providing a systematic approach to assure that individuals possess the appropriate qualifications, certifications, skills, and knowledge to perform assigned functional area activities. Common procedures implement the training process in accordance with INPO Accreditation standards and NRC requirements.

Training on 10CFR50.59 requirements and the PECO Nuclear implementing process is required for personnel who prepare or review 10CFR50.59 Reviews. Activities that have the potential to affect the design bases require a 10CFR50.59 Review.

The Engineering Support Personnel (ESP) training program provides engineering personnel with the knowledge of configuration control processes and design requirements. ESP orientation training includes modules covering configuration control processes and specific design requirements including equipment environmental qualification, single failure, separation criteria, seismic, and ISI/IST requirements.

ATTACHMENT 1

Engineering personnel in the ESP training program must demonstrate their knowledge and ability to perform specific tasks prior to performing them independently. This is accomplished through the use of qualification manuals and the support of mentors and evaluators assigned by management. Qualification manuals for system managers, and for modification, reactor, maintenance/ component, ISI, IST, performance, procurement and regulatory engineers include task qualifications directly related to configuration management and/or design control.

The Engineering Support Personnel Continuing Training Program is designed to maintain, improve, and advance the knowledge and skills of job incumbents. The content of continuing training is determined by a joint committee composed of line organization and training personnel. Program evaluation results, job scope changes, industry and plant events, supervisory needs, procedure changes, and plant system/equipment modifications are used as the basis for determining continuing training topics. Since early 1995, engineering management has addressed personnel performance issues related to design control through the sponsorship of continuing training on topics such as modification process changes, 10CFR50.59 Reviews, UFSAR control, modification testing, and operability determinations.

Operations and maintenance training programs include training on the work processes, administrative requirements, management expectations, and skills necessary to operate and maintain the facilities in accordance with the design bases. The shift manager training program includes several objectives related to managing the plant configuration in accordance with the design bases.

E. Configuration Management Oversight

Configuration Management (CM) oversight activities are in place at PECO Nuclear to identify performance results of design and configuration control processes and implementation activities. CM oversight activities include the performance of multi-disciplined reviews of CM work products; assessments of CM processes and implementation; and development and use of CM performance indicators (PI). The following describes each of these activities.

1. Multi-discipline Reviews

PECO Nuclear has several layers of multi-discipline reviews for changes to its facilities including: Nuclear Review Board (NRB), Plant Operations Review Committee (PORC), and Engineering Quality Achievement Board (EQAB).

The NRB is made up of PECO Nuclear Vice Presidents; the Director, Licensing; the Director, NQA; the Director, Nuclear Engineering Division, and two outside consultants. In accordance with LR-C-13, the NRB reviews 10CFR50.59 Safety Evaluations that identify an Unreviewed Safety Question or involve a change to the Technical Specifications prior to submittal to the NRC. All 10CFR50.59 Safety Evaluations are reviewed under the cognizance of the NRB by the Independent Safety Engineering Group (ISEG) in accordance with NQA-37, "Review of 10CFR50.59 Safety Evaluations."

10CFR50.59 Safety Evaluations for activities that could affect nuclear safety are presented to PORC and approved by the Plant Manager in accordance with A-C-4. The make-up of each PORC is described by the facility's UFSAR and consists of senior plant personnel representing a comprehensive range of nuclear power disciplines.

Engineering Quality Achievement Board (EQAB) assessments are performed by the Engineering Assurance Branch for various engineering programs, processes and modifications and utilize a multi-disciplined team of personnel to perform the assessment. EQAB reviews are performed on various modifications selected by management prior to approving the modification design. The EQAB process is designed as an independent evaluation of the technical, regulatory and business aspects of the area assessed.

2. Assessments

Various assessments of CM processes and activities are performed or requested by PECO Nuclear. These include self-assessments, assessments by NQA, ISEG, and Engineering Assurance, assessments requested by management and assessments by external organizations. Conditions adverse to quality identified under the above assessments are documented and processed as PEP issues in accordance with LR-C-10, Performance Enhancement Program.

Annual self-assessments have been performed at each PECO Nuclear facility since 1990. These assessments include activities which are directly or indirectly related to design and configuration control elements. AG-CG-19, "Self-assessment" provides guidance on performance of self- assessments. Self-assessment open issues are identified and tracked to resolution. Engineering organization self-assessments include many design and configuration control related activities such as modifications, as-building, Engineering Change Request (ECR), calculations and the Design Baseline Document Program.

Corporate and site NQA Divisions perform a significant number of assessments which contain elements that directly or indirectly evaluate design and configuration control activities. These assessments are performed in accordance with NQA-21, "NQA Assessments and Surveillances" The Master Oversight Plan (MOP) identifies the assessment topics, scope and frequency of NQA Assessments. Among the design and configuration control activity areas included in the MOP are Modification and Non-modification Engineering, Modification installation and testing, Temporary Plant Alterations, corporate and station Document Control, and vendor manual control. Assessments of these and other areas are utilized in part to evaluate various aspects of configuration management such as design control, installation, testing, documentation, plant operation and configuration.

An assessment process in compliance with ANSI N45.2.12 and implemented through NQA-21, is utilized to perform LGS and PBAPS Technical Specification required assessments. These assessments are performed under the cognizance of the Nuclear Review Board. Additionally, as part of the NQA Department, the ISEG at LGS and PBAPS performs reviews of CM activities.

The Engineering Assurance Branch of the Nuclear Engineering Division provides Design Authority oversight assessment for internal and delegated design activities. These assessments review configuration management elements and include an integration of technical assessments, data reviews, process reviews and PEP corrective action follow-up evaluations. Input for assessments is obtained from monitoring Performance Indicators, NQA assessments and surveillances, and functional area self-assessments.

PECO Nuclear performs a periodic corporate assessment of the facilities using the INPO/WANO assessment criteria. These assessments are generally performed on a two year frequency. Assessment reports are prepared and distributed to appropriate management and recommended corrective actions are initiated.

PECO Nuclear is regularly assessed by non-PECO Nuclear personnel under the Joint Utility Management Audit (JUMA) program as well as other industry peer evaluation programs.

ATTACHMENT 1

3. Performance Indicators

Performance indicators (PIs) are utilized to track and monitor various design aspects of Configuration Management performance. Some examples of areas monitored by PIs include drawing as-building timeliness and quality, temporary plant alteration (TPA) status, Licensee Event Report (LER) causes and results, maintenance backlog and aging, and Nonconformance Report (NCR) aging. These PIs are some of the many which are reviewed monthly by senior management. Adverse trends and performance below expectations are thus made apparent for management to initiate appropriate action.

III. CONFIGURATION MANAGEMENT OVERSIGHT RESULTS

A. Multi-discipline Review Results

The in-line multi-discipline reviews have provided effective barriers and also have helped identify opportunities to improve. PORC reviews of modifications, procedure revisions, and Safety Evaluations are challenging and provide an in-depth analysis of the important issues. Both the LGS and PBAPS 1995 self-assessments identified PORC reviews as a strength. NRC Inspection Report PBAPS 95-80/80 noted that "PORC maintained a critical attitude toward safety issues." LGS 95-80/80 contained similar language. LGS NRC Inspection 96-09/09 noted that PORC members "actively participated in meetings with open discussions on the plant issues while maintaining a focus on safety."

B. Assessment Results

Oversight activities in the areas of design control and configuration control during the past two years indicate that PECO Nuclear generally performs work to high standards as demonstrated by above average SALP ratings. Nevertheless, instances of performance weakness have been identified by both internal and external assessments.

The internal assessment results discussed below demonstrate that PECO Nuclear's self-assessment standards are high, and that a questioning and critical approach is taken during assessment activities, with the main goal being to continually improve performance. By their nature, internal assessment reports often focus on the problems identified during the assessment, rather than the strengths. The following assessment results discuss several cases of performance weakness. Most of these weaknesses are self-identified and reflect PECO Nuclear's commitment to critical oversight.

1. NQA Assessments and Surveillances

NQA assessments and surveillances of CM activities have identified that engineering performance in the design of modifications is generally good. These assessments are effective in identifying improvement opportunities. A 1995 NQA assessment evaluated the adequacy of the design change process implemented by both the Nuclear Engineering Division and the stations' Engineering Divisions. The assessment team judged the MOD Teams to be technically strong, that walkdowns are sufficiently detailed and consistently performed, and that modification packages adequately address all major configuration documents. However, weaknesses in configuration management and design review were identified. In particular, three PBAPS category A1 drawings were not updated within the required time frame. This occurred because the ECR that was supposed to be incorporated contained conflicting information that required resolution. The ECR that was generated to resolve the conflicting information was not processed in a timely manner and, when it was, it too contained erroneous information. For both the original and follow-up ECRs, the assessment report also cited the independent reviewer of the

ECR for not identifying the problems and having them corrected. PEP issues were initiated to address these weaknesses and corrective actions have been implemented.

A LGS 1995 NQA assessment evaluated the adequacy of modification installation activities at LGS. The same strengths identified by the above assessment were noted. Some weaknesses were noted in the field installation activities and were tracked by PEP issues. Another NQA assessment performed in the third quarter of 1996 assessed the implementation of the revised modification process. No deficiencies were identified in the five sampled MOD ECRs. However, a weakness in the training of personnel involved in the new modification process was identified and corrective action development is underway by a task team.

During the first quarter of 1996, an integrated assessment was jointly performed by NQA assessment personnel and ISEG personnel. This assessment evaluated design control activities for the PBAPS modification which replaced the Unit 3 Main Steam, HPCI, and RCIC Leak Detection Systems. Although satisfactory performance was noted in a number of areas, several weaknesses were identified; the most significant being a deficiency in the design of the electrical circuits to satisfy the single failure criterion. The assessment revealed that the modification team lacked sufficient understanding of the single failure criterion and the electrical separation criterion. Other areas of concern were noted with the preparation and use of the Design Input Document and determination of acceptance test criteria. It was also noted that the 10CFR50.59 Review did not completely evaluate all failure modes of the revised design. PEP issues were initiated as a result of this assessment and some corrective actions remain to be completed.

2. ISEG Reviews

ISEG reviews of engineering activities have consistently resulted in improvements and corrective actions. As an example, the PBAPS 1994 Annual Summary Assessment Report (ASAR) discusses a configuration management weakness concerning procedures and drawings not being revised in a timely manner after installation of a modification. Corrective actions resulting from the initial ISEG report, as well as from Nuclear Engineering Division self-assessment results, have significantly improved the timeliness of as-building of drawings throughout the second half of 1995 and all of 1996. The former backlog of overdue as-builds has been eliminated and the monthly performance indicator that tracks as-building performance identifies that drawing updates are currently well managed.

NRC Inspection Report PBAPS 95-80/80 documents the NRC's review of PBAPS ISEG activities during the period from May 1994 through June 1995. Twenty-six ISEG reports issued during this period were reviewed, including the 1994 ASAR discussed above. The NRC report stated that "the inspectors found the reports to be thorough, challenging, and of good quality," and also "concluded that ISEG was performing effective independent assessment of the Peach Bottom organization." NRC Inspection Report LGS 95-80/80 also states that LGS "ISEG performed critical assessments of appropriate activities and events" and that the twenty-five 1994 "ISEG assessments demonstrated a strong safety perspective with the safety significance clearly articulated."

3. Corporate Assessments

During January 1996, a PECO Nuclear Evaluation of PBAPS was performed using the evaluation criteria of "Performance Objectives and Criteria for WANO Peer Reviews," Rev. 1. Overall station performance was determined to be strong; however, an area for improvement was identified for modification testing. It was noted that although the modification testing requirements for two modifications were acceptable, the depth of testing detail varied considerably. It was recommended that post installation testing requirements continue to be monitored.

During November 1996, a PECO Nuclear evaluation of LGS was performed using revised INPO performance criteria. Overall station performance was determined to be strong, as measured by high capacity factors, short and safe refueling outages, low radiation exposure, low industrial safety accident rates, and low cost of generation. The assessment team identified an opportunity for improvement in the area of plant status and configuration control. Minor deficiencies were identified with temporary plant alterations and other plant configuration control issues. Corrective action for these issues is in progress.

4. Industry Peer Assessments

The most recent Industry Peer Review of LGS was performed in September 1995. In the area of Engineering Support, a weakness was identified in the modification testing and closure of design packages. Inadequate modification testing of the hydrogen recombiners and a recorder which provides drywell and suppression pool pressure indication were cited. In addition, three drawing and procedure update omissions following modification closure were cited. In the period since this assessment, significant modification process improvements for modification testing and design package quality have been made and performance expectations in these areas have been communicated to appropriate engineering personnel.

An Industry Peer Review of PBAPS was performed in March 1996. In the area of Engineering Support, no weaknesses or areas for improvement were identified.

PECO Nuclear requested an Industry assist visit in the area of configuration management during August of 1995. The scope of the visit was to review CM activities, including transition plans for implementing an enhanced modification process. Strengths noted during the review include (1) the release based process change implementation methodology results in the coordination of procedure revisions, computer software upgrades, and staff training to support implementation on the release date; (2) automated recording, tracking, and updating of changes to engineering document records allows personnel at all sites to quickly determine change and base document status; and (3) the staffs are knowledgeable of their current responsibilities, cognizant and supportive of upcoming changes, and recognize that the document databases are critical to effective CM. Suggestions for improvement include (1) identification of an opportunity to increase the effectiveness of the modification process transition plan; (2) flowcharting the engineering modification process so redundant requirements, overlapping requirements, and non sequential steps become more clearly highlighted for evaluation; and (3) evaluation of the as-built document types or categories to identify consolidation opportunities. The suggestions for improvement have been evaluated and appropriate actions taken.

5. Engineering Quality Achievement Board (EQAB) Assessments

EQAB assessments have identified generally good, but inconsistent, engineering performance in the design of modifications. EQAB reviews are performed on various modifications selected by management prior to approving the mod design. EQAB assessments performed during 1996 identified improvement opportunities for the modification packages reviewed. Although each modification's design output documents were found to adequately address and implement the design bases, numerous drawing errors and instances of insufficient engineering detail were identified. Each EQAB assessment documents the resolutions to the assessment findings. Programmatic recommendations are also provided where appropriate and tracked by PIMS Action Requests. EQAB assessments are distributed to the appropriate levels of engineering management and the NRB. Engineering Assurance prepares quarterly reports that summarize the issues identified by EQAB assessments during the period and distributes them to line management and the NRB.

6. Multi-discipline Self-Assessments

During 1994 and early 1995, several PEP issues identified an apparent weakness in the electrical engineering portion of recently installed modifications. In May 1995, a multi-discipline team consisting of eight experienced engineers and designers was charged by PECO Nuclear management to perform a special electrical work products assessment that focused on the technical quality of the electrical portions of selected, recently designed modifications. Modification design documentation deficiencies were identified and corrective actions initiated. No nuclear safety significant issues were identified during this assessment. Generic corrective actions that resulted from this assessment include programmatic enhancements, reinforcement of technical quality expectations, and minor procedure revisions.

7. NRC Inspections/Assessments

An indicator of each station's overall performance is provided by the NRC's Systematic Assessment of Licensee Performance (SALP) program. The most recent SALP Report for LGS (May 1995) rated the Engineering functional area as Category 1, with modification and design work cited to reflect high technical quality. The Maintenance and Operations functional areas also received category 1 SALP ratings.

The most recent SALP Report for PBAPS (December 1995) rated the Engineering functional area as Category 2, citing generally good, but inconsistent, engineering performance. The design, planning, and implementation of modifications were characterized as usually good; however, occasional lapses in the quality of modification and other technical work had occurred during the evaluation period. The previous SALP rating for the PBAPS Engineering functional area was also Category 2. The Maintenance and Operations functional areas received category 1 SALP ratings.

NRC Inspection Reports received since the latest SALP reports indicate that engineering performance remains generally good; however, occasional lapses in performance continue to occur. NRC Inspection Report LGS 95-12/12 and its accompanying Notice of Violation provides a recent example of a lapse in engineering performance, although one of the cited deficiencies dates from a 1989 modification. The violation addresses inadequate design controls for modification to three primary containment hydrogen recombiners, including an inadequate modification Acceptance Test Plan and post modification testing. Corrective actions have been implemented for this issue.

This same NRC Inspection Report (LGS 95-12/12) identifies effective engineering performance in the area of 10CFR50.59 Reviews. The inspector found the procedures governing 10CFR50.59 Reviews to be complete, to provide adequate details, and to be consistent with the regulation. Ten 10CFR50.59 Reviews were selected from a group of 114 and the inspector concluded that PECO Energy had adequately implemented the requirements of 10CFR50.59 for this selected sample.

The two NRC Inspection Reports identified above which reported adequate ISEG performance (PBAPS 95-80/80 and LGS 95-80/80) primarily addressed PECO Nuclear's problem identification and resolution processes and self-assessment activities. At both stations the problem identification and corrective action processes were judged to be effective. At Limerick, self-assessment was determined to be excellent, and at Peach Bottom, a strong dedication to self assessment by PECO management was observed as well as numerous examples of successful self-assessment activities.

A recent Notice of Violation which addresses modification activities was identified by NRC Routine Integrated Inspection Report PBAPS 96-06/06. The inspection report cites weak analysis of a design issue dealing with the time delay start of an RHR pump in response to a loss of coolant accident with the associated emergency diesel generator in test mode. The NOV cites that PECO Nuclear did not fully understand how a modification affected other important safety systems. Recently, PECO Nuclear has provided a response to the NRC regarding this NOV which describes the corrective actions that have been completed in addition to the corrective actions still in progress.

C. Performance Issues

Although many assessments performed since the PBAPS Restart and LGS Unit 2 Start-Up time frame report that PECO Nuclear's CM processes are adequately implemented, three areas have had occasional performance problems more often than others. These areas are performance of 10CFR50.59 Reviews, the use of Design Input Documents, and establishing comprehensive acceptance tests for modifications. A discussion on each of these areas follows.

1. 10CFR50.59 Reviews

NQA, ISEG, and the NRC have each identified various problems in the area of 10CFR50.59 Reviews. Even though corrective actions have been implemented for each issue, problems are still occasionally identified in this area. To some extent, the continuance of cited problems in this area is partially attributable to higher expectations of performance and a broader, and more experienced understanding of the philosophy and requirements of 10CFR50.59 by the internal assessment organizations. Following are examples of problems experienced in the area of 10CFR50.59 Reviews. Process improvements in this area are discussed below.

a) Assessment Observations

One example of an internal assessment that discusses performance problems in the area of 10CFR50.59 Reviews is documented by a 1994 PBAPS ISEG Report 94-09. ISEG concluded that there was less than adequate understanding of what constitutes the SAR and/or which procedures are described in the SAR. A new procedural document, LR-CG-13, "Preparing 10CFR50.59 Reviews," provides corrective action to this problem by providing additional discussion on what constitutes the SAR as well as the meaning of the phrase "procedures as described in the SAR." ISEG also reported that many of the 10CFR50.59 Determinations reviewed provided less than adequate bases for the answers to the four Determination questions and that the reviews of the UFSAR appeared to be incomplete.

Another 10CFR50.59 problem area that has been identified by ISEG, NQA, and the NRC is that sufficient detail is not consistently provided to fully identify the scope of the change being evaluated or to support the basis for the response to either the Determination questions or the Safety Evaluation questions. For example, a 1995 LGS ISEG report discusses an ISEG review of the 10CFR50.59 Safety Evaluation supporting the disposition to NCR LG 94-00014 (subject: inadequate voltage at the ECCS Inverters) and related calculation LE-0069. ISEG reported that an adequate basis to support the 10CFR50.59 Safety Evaluation's conclusion was not documented by the Safety Evaluation. Following a discussion with the preparer of the Safety Evaluation, ISEG did concur with the Safety Evaluation's conclusion that an Unreviewed Safety Question did not exist. As a result of the

ATTACHMENT 1

ISEG review, the Safety Evaluation was revised to include the appropriate level of detail necessary to support the conclusion.

A recent NRC Routine Integrated Inspection of PBAPS activities (reference NRC Inspection Report PBAPS 96-06/06, dated 10/10/96) resulted in a Notice of Violation that cited failure to implement 10CFR50.59. The deficiency was originally identified by NRC Inspection Report PBAPS 95-27/27, dated 01/30/96. Specifically, PECO Nuclear was cited for operating the Standby Gas Treatment (SBGT) system differently than described in the UFSAR without preparing or maintaining a written 10CFR50.59 Safety Evaluation. It was noted that an unanalyzed system configuration was allowed by the SBGT system operating procedures such that the system could not meet single failure criteria. Although this cited deficiency is recent, the root cause of the problem dates back to the original preparation of the system operating procedures (see Attachment 1 response to request b) Section I.C.2.e) for additional information).

The Performance Enhancement Program requires that when problems are identified as repetitive, the rigor with which PECO Nuclear determines the root causes of the problem is intensified. The repetitive nature of the problem is also raised to increasingly higher levels of management. LR-C-10 requires trending of performance issues and identification of repetitive problems by the Experience Assessment Coordinators. PECO Nuclear is confident that the requirements for 10CFR50.59 Reviews are captured by the controlling procedures. Performance problems are processed in accordance with LR-C-10, and repetitive performance problems result in an escalation of corrective actions to ensure recurrence is precluded.

A PEP issue was issued to track an observed adverse trend in the frequency of 10CFR50.59 Review problems being identified by oversight organizations rather than by the line organizations. Between September 1993 and December 1996, 90 PEP issues have been initiated which identify problems with 10CFR50.59 Reviews. These PEP issues have been categorized into the following five problem categories: content deficiencies, missing 10CFR50.59 Reviews or not performed, improper disposition of the 10CFR50.59 Review after it was written, lead-in process results in no 10CFR50.59 Review being generated, and governing procedure related. An ongoing evaluation of performance in the area of 10CFR50.59 Reviews is being managed by this trend PEP issue. Although PECO Nuclear recognizes that continued management attention is needed in this area, the trend results of the data indicate that performance is improving. During December 1996, ISEG performed a review of recent 10CFR50.59 Reviews and reported that the quality of 10CFR50.59 Reviews has improved during the past year.

b) Process Improvements

PECO Nuclear has committed considerable resources to improving performance in the area of 10CFR50.59 Reviews. During 1995, it was recognized that clarification was needed to identify those documents in addition to the UFSAR that should be considered when preparing or reviewing a 10CFR50.59 Review. Preparers and reviewers of 10CFR50.59 Reviews reported that LR-C-13, 10CFR50.59 Reviews, was too long, making it difficult to find specific requirements and guidance within it, and that some of the documents referenced by the UFSAR that should be considered when preparing or reviewing a 10CFR50.59 Review were difficult to obtain.

Another improvement initiative has been implemented by provision of Personal Librarian Software (PLS), a computer database search program which serves as a repository for many of the documents that should be considered when preparing or reviewing a 10CFR50.59 Review.

In 1996, the 10CFR50.59 Review process was further improved. The procedure LR-C-13 was split into two documents: LR-C-13, and LR-CG-13. The revised LR-C-13 establishes the process requirements, states management expectations, provides an overview of the purpose of the regulation and the SARs, identifies those documents to be considered when preparing or reviewing a 10CFR50.59 Review, and clarifies what is meant by procedures described by the SARs. LR-CG-13, Performing 10CFR50.59 Reviews, guides the preparers step-by-step through the process of preparing and documenting a 10CFR50.59 Review. Training on the revised 10CFR50.59 process was again provided as part of the roll-out of these revised procedures. Initial user feedback has been positive. The effectiveness of these revisions is still under evaluation.

In summary, multi-discipline reviews and internal assessments are identifying occasional performance problems with 10CFR50.59 Reviews. PECO Nuclear's multiple layers of in-line reviews and follow-up assessments and process improvements attest to PECO Nuclear's determination to achieve consistently high performance in this area.

2. Design Input Documents

Deficiencies related to the thoroughness of Design Input Documents for modifications have occasionally been identified by various assessments. As a recent example, in February 1996, a PBAPS Integrated Mod Assessment was performed for the replacement of the Unit 3 Main Steam, HPCI, and RCIC Leak Detection Systems modification. This assessment identified that clarification was needed on management's expectation for using Design Input Documents (DIDs) when making modification design decisions, accomplishing engineering verification, evaluating design changes, and writing plant procedures. As corrective action a new section of the MOD Manual, MOD-CM-1, was issued which defines management's expectations and provides examples. This new guidance was provided in May 1996.

3. Modification Acceptance Tests

PECO Nuclear's Performance Enhancement Program requires evaluation of events to determine root causes and generic implications of the events. As a result of a PEP Issue which addressed a contact configuration error for a diesel generator relay, one of the corrective actions to address the identified shortcomings in the modification testing process was the creation of the supplemental review process that directly supports the goals of assuring that the change has been correctly designed, completely tested and conforms to the evaluated configuration. The supplemental review was developed to improve the overall quality of modification packages. In this process, appropriate design changes go through a screening step to identify those having a level of complexity warranting the supplemental review after design, installation planning, and acceptance test development are completed. The process includes a review to assure that all components affected by the design change are tested and that design goals are met.

D. Performance Indicator results

Each CM functional area has some PI coverage. Management utilizes PIs to assess performance and to target areas for improvement. As examples of existing PIs, two typical indicators included in the management monthly update reports are drawing as-building quality and timeliness, and NCR status. Over the past one to two years, drawing as-building has shown improved performance in timeliness while maintaining high quality. NCR closure timeliness has also improved thus reducing the overall age and backlogs of NCRs. A project to further identify and improve PIs to enhance their coverage and effectiveness has been initiated.

IV. SUMMARY CONCLUSION

PECO Nuclear has adequate engineering design and configuration control mechanisms in place to ensure that the design bases are maintained. These mechanisms are an integration of procedural, programmatic, and management controls which provide reasonable assurance that:

- The plants conform to approved design requirements.
- The physical and functional characteristics of the plants are accurately reflected in controlled documents.
- The status of design, plant and functional design changes, temporary plant alterations, and associated documents are readily accessible to appropriate line organizations.
- The Design Bases are accurately maintained.

Changes to the UFSAR are made under a common Engineering Change Request (ECR) process. 10CFR50.59 Reviews are required to be prepared for both changes to the facility as described in the SAR and changes to procedures as described in the SAR. Adequate processes are in place to allow for identification, review, approval, and capture of necessary UFSAR revisions in support of the UFSAR annual update and in accordance with 10CFR50.71(e) requirements. Changes to other licensing bases documents also require preparation of 10CFR50.59 or 10CFR50.54 Reviews prior to submittal to the NRC in accordance with applicable regulations. Appropriate configuration control elements are embedded in policies, directives, and procedures of PECO Nuclear functional areas that interface with the plant design and the plant design bases. The common procedure preparation and revision process requires that they be evaluated by a 10CFR50.59 Review. This process also requires a review that the procedure adequately implements requirements of source documents including the SAR, non-SAR commitments, and policies and directives. In addition, PECO Nuclear's CM functional area procedures and processes implement applicable 10CFR50 Appendix B criteria. Adequate CM oversight processes are in place to monitor compliance with commitments, regulations, and approved procedures. Performance problems which identify Conditions Adverse to Quality in the area of design and configuration control are reported in accordance with the Performance Enhancement Program. This program requires appropriate rigor for performance of root cause analyses, determination of generic implications, execution of corrective actions, and analysis of the effectiveness of the corrective actions.

PECO Nuclear has concluded that adequate processes are in place and their implementation is monitored to reasonably assure configuration management of the SAR, design bases, and design documents.

ATTACHMENT 1

REQUEST (b)

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

RESPONSE

This response provides PECO Nuclear's rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures. Section I addresses the consistency between the existing station procedures and design bases requirements. This consistency can be challenged whenever the design is altered or procedures are revised. Section II discusses the PECO Nuclear procedure revision process and the controls in place to maintain the procedures consistent with the design bases and provides an evaluation of the process and its implementation. Section III discusses the processes by which PECO Nuclear changes the stations' design bases and how these processes assure procedures are revised to remain consistent with the design bases and includes an evaluation of the processes and their implementation. Section IV provides a summary conclusion.

I. ACCURACY OF PROCEDURE CONTENT:

PECO Nuclear has undertaken a number of improvement projects and assessments during the last ten years which included, to varying degrees, review and revision of station procedures. These activities provided multiple opportunities to review station procedures against design bases requirements. Although no single activity has been performed to provide a 100% verification of procedure content against the design bases, each activity has achieved incremental improvements in the technical consistency and accuracy of the LGS and PBAPS station procedures. The major activities include:

A. Projects:

1. Projects Common to LGS and PBAPS:

a) Development of Design Baseline Documents (DBD):

PECO Nuclear undertook a project from 1990-1995 to develop system and topical DBDs. A total of 152 DBDs were developed. DBDs were created for all safety-related systems, systems important to safety and systems important to efficient plant operation. Topical DBDs were developed for areas such as Environmental Qualification, Fire Safe Shutdown, and Regulatory Guide 1.97, Post Accident Monitoring. DBDs were created to capture the system or topical licensing and design bases in a single document to aid personnel in performing plant design changes and 10CFR50.59 Reviews. The DBDs address system functions, alignments, controlling parameters, and design features for normal, abnormal, and accident conditions. The DBDs derive information from design documents such as drawings, specifications, and calculations.

The development process for each system DBD included a Testing Validation Report. The report identified if a test had been performed for each controlling parameter and determined if the acceptance criteria values contained in the test encompassed the parameter value. Controlling parameters are the specific numerical values chosen as reference bounds for system design. Controlling parameters are associated with all system functions. Examples of controlling parameters are: temperature, pressure, level, voltage, and current.

ATTACHMENT 1

In order to develop the Testing Validation Report, 20-30 station procedures per system were reviewed. In cases where the controlling parameters were tested during the startup test program and are not tested on an ongoing basis, the startup test procedure acceptance criteria values were reviewed. Action items were initiated and dispositioned in PIMS to address discrepancies identified between the station test procedures and the DBD values.

The DBDs provide a consolidated source of system and topical area design bases requirements for personnel developing or revising station procedures. DBD Testing Validation Reports verified, where applicable, that the acceptance criteria values in station test procedures encompass the DBD controlling parameter value.

b) Limerick and Peach Bottom Power Rerate Projects:

PECO Nuclear conducted a unit rerate project from July 1992 to February 1996 to increase the core thermal power for each LGS and PBAPS unit (total of 4 units) by 5 percent. This project required that station procedures be reviewed for potential impacts caused by the changes in station operating parameters due to unit rerate. Approximately 1000 LGS and PBAPS procedures were revised to incorporate the new unit rerate operating parameters.

LGS and PBAPS station procedures w... reviewed and revised as required to assure that unit rerate design data was incorporated into the station procedures. 10CFR50.59 Reviews were performed for the revised procedures.

c) Common Maintenance Procedure Project:

PECO Nuclear undertook from 1990 to 1995 an effort to consolidate and make common approximately 680 maintenance procedures and guidelines to improve effectiveness of the work force. Maintenance procedures common to both plants were developed utilizing the best methods and practices from both stations while recognizing the unique technical and regulatory requirements at each station. Development of these procedures required the procedure author to review design documents such as specifications, equipment vendor manuals and equipment qualification reports to assure the procedures met the necessary technical requirements. The procedures were processed through the procedure revision process including the preparation of a 10CFR50.59 Review for each procedure.

d) Peach Bottom/Limerick Transient Response Implementation Plan (TRIP) Procedure Emergency Procedure Guidelines (EPG) Rev 4 Upgrade:

This activity involved incorporation of the BWR Owner's Group (BWROG) EPGs, Rev. 4 into the Peach Bottom and Limerick emergency operating procedures (EOP). The upgrade resulted in a re-write of all the TRIP procedures. During the re-write, PECO addressed several PBAPS specific implementation discrepancies in the existing procedures which were identified in NRC Inspection Report 88-200/200. The new procedures became effective in the 1990-1991 time frame.

Generation of the procedures was performed using a rigorous process that documented a step by step justification of the plant-specific version of the generic EPG step. This involved obtaining design information from controlled plant documents and revising existing calculations and performing new calculations using approved methods. A 10CFR50.59 Review was prepared against the Safety Analysis Report (SAR) including the Updated Final Safety Analysis Report (UFSAR) and the Safety Evaluation Report (SER) associated with revision 4 of the EPG's.

ATTACHMENT 1

The following aspects of the EPG Rev. 4 - TRIP Procedure Upgrade developed the link between the revised procedures and design basis information contained in the SAR or in other design basis documents:

- Design basis information from controlled plant documents was used when developing station-specific procedure steps from the generic BWROG guidance
- The TRIP Procedure verification and validation process ensured fidelity between the procedures and regulatory commitments
- The procedure review and approval process required performance of 10CFR50.59 Reviews

e) PECO Nuclear Response to Generic Letter 96-01, Testing of Safety-Related Logic Circuits

In response to GL 96-01, PECO Nuclear is reviewing surveillance procedures to ensure they adequately test safety-related automatic actuation logic circuitry so that a failure of an essential electrical component (e.g., relay contact) will not be undetected for an extended period of time.

A thorough review of the licensing bases of Peach Bottom and Limerick will be completed to identify those systems, structures, and components that are included in the scope of GL 96-01. A review of one logic division of each of the systems within the scope of GL 96-01 will be completed. Limerick reviewed eleven systems. Ten systems were reviewed with no deficiencies found which would result in a system not performing its safety function. The review of the 4 kV system identified four issues regarding the logic testing. Four NCRs have been initiated to develop the appropriate corrective actions. A PEP issue has also been initiated to evaluate generic implications. As committed to in the response to GL 96-01, PECO Nuclear will provide written notification to the NRC within 30 days of the completion of this review.

f) Historical Commitment Backlog Review Effort/Commitment Annotation Program

During the implementation of the Commitment Tracking Program in 1988, it was identified that some commitments issued prior to 1988 may not have been properly implemented in PECO Nuclear programs. Further, continued compliance with commitments involving NRC, INPO, and ANI could not be easily verified because prior to 1988, PECO Nuclear did not have a formal commitment tracking program.

The Historical Commitment Backlog Review Effort located and evaluated several thousand documents issued or received by PECO Nuclear since commercial operation (i.e., 1974 to 1988 for PBAPS Units 2 and 3; 1986 to 1988 for LGS Unit 1). The final screening resulted in the annotation of over 1,000 commitments in PECO Nuclear procedures/programs. No significant non-compliances with these historical commitments were identified by PECO Nuclear during the disposition phase of the review effort.

The scope of the Commitment Annotation Program includes annotation of implementing documents with programmatic commitments (i.e., ongoing actions) made by PECO Nuclear to external organizations, and annotation of implementing documents with significant programmatic corrective actions associated with the review of industry operating experience (OEAP) and significant PEP issues. The Commitment Annotation Program provides

ATTACHMENT 1

reasonable assurance that commitments/corrective actions implemented by PECO Nuclear are properly maintained.

In 1992 and 1993, as a separate task of the Historical Commitment Backlog Review Effort, Limerick performed a review of selected UFSAR commitments to verify the commitments were incorporated in Station procedures. Sections of the LGS UFSAR which contained specific commitments regarding plant operation or specific commitments to a document, such as a Regulatory Guide or Industry Standard, were identified. Examples of commitments regarding plant operation include: inspection of systems during normal operation to ensure minimal leakage, or requirements for maintaining valves closed by administrative means during certain plant conditions. Once identified, these UFSAR commitments were reviewed for accuracy and station procedures were reviewed to ensure the item was incorporated. Approximately 400 UFSAR commitments were verified to be implemented in station procedures.

The Commitment Annotation Program provides assurance that commitments/corrective actions implemented by PECO Nuclear are properly maintained. The program provides value by ensuring that personnel do not unknowingly remove programmatic commitments/corrective actions from procedures/ programs and thus expose PECO Nuclear facilities to non-compliance with external regulations and requirements or to the potential for repeat events at the stations. In addition, at Limerick approximately 400 UFSAR commitments were verified to be implemented in station procedures.

2. LGS Projects:

a) Procedure Partnership

In 1994, in response to concerns relative to human factoring, Limerick performed a comprehensive re-write of all System Operating (S), Off Normal (ON), Operational Transient (OT), Event (E), Special Event (SE), and TRIP (T-200) procedures. The purpose of the re-write project was to assure that the procedures listed above were the highest quality possible, technically accurate, efficient, and user friendly. The project was performed by establishing partnerships between system managers from Site Engineering and Operations personnel. The newly prepared procedures were walked down to ensure accuracy with respect to plant configuration typically using the system Piping and Instrument Diagrams (P&IDs).

This project resulted in the re-writing of 1200 procedures. Each procedure was assured to be technically correct using current references. Inconsistencies between technical documentation and plant configuration were resolved using the appropriate corrective action process. 10CFR50.59 Reviews were performed for the revised procedures.

b) Surveillance Test (ST) & Routine Test (RT)-Procedure Re-write Project:

In 1995, Limerick completed a total re-write of all surveillance and routine tests used by Operations personnel. The purpose of the project was to re-write Operations Surveillance Test (ST)/Routine Test (RT) procedures to be efficient, user friendly, and technically accurate. Partnerships were set up between Site Engineering system managers and Operations personnel. As with the procedure project completed in late 1994, the procedure partners were given newly human-factored procedures and instructed to assure the following:

- the procedure was technically correct
- sequence was logical
- all expected plant responses including alarms and process changes were identified
- component identification matched labeling
- human performance considerations—for example, traveling from location to location were considered

A walkdown of the procedure was conducted when possible to assure the above expectations were met.

When the partners completed each procedure, the procedure was processed through the procedure approval process.

Approximately 600 procedures were reviewed by both a system manager and an operator to assure they were accurate. Additionally, where possible, each procedure was walked down to assure component identification matched plant labeling. 10CFR50.59 Reviews were performed for the revised procedures.

3. PBAPS Projects:

a) PBAPS Operating Procedure Rewrite:

As a result of both human factor and technical inadequacies in station operating procedures (which include System Operating and Abnormal Operating procedures) a procedure rewrite project began in 1987 to revise approximately 2000 procedures. The Procedure Writer's Guide used for the project discussed the technical information to be collected to be used to develop the procedures. The source documents included engineering drawings, vendor technical manuals, Technical Specifications, and commitments to regulatory agencies.

System Operating and Abnormal Operating procedures were rewritten using a process which included the following requirements which provided a link between the revised procedures and design bases information contained in the UFSAR or in other design and licensing baseline documents:

- use of current, controlled engineering drawings and prints
- review of Tech Spec requirements and applicable regulatory commitments
- performance of Verification and Validation check lists, which included a review of Inservice testing (IST) criteria and plant walkdowns
- performance of 10CFR50.59 Reviews

b) PBAPS Maintenance Procedure Re-write Project:

A 1988 Industry evaluation report concluded that procedures did not exist for many maintenance activities and that existing procedures were formatted allowing the potential for human performance errors. Subsequently, all maintenance procedures (M) were replaced by new procedures; and new site-specific instrument and control procedures (IC) were written to replace the existing Testing and Lab procedures. The upgrade program was completed during the summer of 1991.

The upgrade program plan required procedure writers to incorporate information contained in prints, vendor manuals, commitments, regulations, and equipment history. A walkdown was performed for each new procedure to verify as-built configuration, nameplate data, and equipment labeling. Upgraded procedures received 50.59 determinations and safety evaluations as appropriate. Many of the upgraded procedures have been revised since 1991, with some being replaced by procedures common with LGS.

All Maintenance and I&C procedures were rewritten following administrative controls for procedure review and approval which included the following requirements which provided a link between the revised procedures and design basis information contained in the SAR or in other design basis documents:

- use of current, controlled engineering drawings and prints
- review of Tech Spec requirements and applicable regulatory commitments
- review of draft procedures by an Environmental Qualification subject matter expert
- performance of 50.59 determinations, and, as appropriate, Safety Evaluations.

c) PBAPS Improved Technical Specification Project:

This project converted the custom PBAPS Technical Specifications (Tech Specs) to Improved Tech Specs (ITS) in accordance with the standard issued as NUREG-1433. One of the main project objectives was to align the Tech Specs directly with the design bases. This resulted in Limiting Conditions for Operations (LCOs), Tech Spec Actions (TSAs) and Surveillance Requirements (SR's) being added, deleted, and modified. To implement these changes approximately 3000 procedures were revised. These revisions could be as simple as changing references, or as complex as developing new logic system functional tests. This effort assured that all new or revised Surveillance Requirements are performed and that the acceptance criteria in the procedures comply with the Surveillance Requirement criteria. The ITS Project began in February 1993 and became effective January 1996.

The PBAPS ITS Project performed a comprehensive review that verified that Technical Specification Surveillance Requirements and applicable regulatory commitments were identified and incorporated into station surveillance tests and procedures. Design bases information, including the UFSAR, calculations, and docketed letters, was used extensively in revising LCOs, TSAs and Surveillance Requirements. The revised procedures, therefore, are linked to design bases information contained in the SAR, other design bases documents, and Improved Tech Specs. Additionally, the ITS Bases was completely rewritten and now contains more highly detailed design information. The ITS Bases is a valuable resource for maintaining and revising procedures.

d) PBAPS Surveillance Test Re-write Project:

The PBAPS ST Re-write Project (1990 - 1992) encompassed existing surveillance tests, tests associated with in-service inspection, tests associated with the Fire Protection plan, and routine tests. The re-write included a screening process to identify surveillance requirements contained in Tech Specs and other regulatory commitments. The verification and validation program included reviews by system engineers and end users. The procedure approval process required 10CFR50.59 Reviews; therefore, the procedures were reviewed against the SAR.

The project performed a comprehensive review which assured that Technical Specification surveillance requirements and applicable regulatory commitments were identified and incorporated into station surveillance tests. Since Technical Specifications implement many design bases requirements, this program established a link between the revised test procodures and these design bases requirements. Additionally, the project's review and approval process required: (1) the use of current, controlled engineering drawings and prints, and (2) the performance of 10CFR50.59 Reviews.

B. Assessments:

1. Assessments Common to LGS and PBAPS:

a) Safety System Functional Inspection (SSFI):

PECO Nuclear conducted SSFIs on six (6) safety systems at PBAPS and four (4) safety systems at LGS. System selection was based on importance to plant safety and recognition of prior system concerns. The SSFIs were conducted during 1990 to 1993 on the following systems:

PBAPS:

- High Pressure Service Water System (HPSW)
- Standby Gas Treatment System
- Feedwater System
- Service/Instrument Air and Nitrogen Systems
- 250V/125VDC Systems
- Diesel Generator - AC Emergency Electrical System

LGS:

- Emergency Service Water (ESW)
- Feedwater System
- High Pressure Coolant Injection System (HPCI)
- Diesel Generator and AC Emergency Electrical Systems

SSFI's employed a "deep vertical slice" methodology and team interaction techniques developed as detailed in NSAC 121, November 1988. The systems were reviewed in substantial technical depth to evaluate design and modification processes and to assess implementation of the design in operations, maintenance, testing, and training. The inspection was conducted by an independent team of SSFI experienced contractors with the assistance of PECO Nuclear's NQA organization.

ATTACHMENT 1

The SSFI Review Plans required review of plant procedures for consistency with the design bases documents. Specifically, the plans required:

- a review of UFSAR sections, design bases documents, and drawings for correlation of design with operating procedures

- a review of surveillance procedures to assure that surveillance requirements are adequately implemented and reflect actual system and component functions and the design intent

- a review of maintenance procedures and maintenance test procedures to identify weaknesses and inconsistencies with the design intent for the component or system function

- a review of selected modifications to assess whether changes that affect the system operation and maintenance have been adequately addressed in station procedures

Each SSFI involved the review of at least fifty (50) station procedures. Each inspection team concluded that the selected system was capable of performing its safety function under postulated design basis accident conditions. The four SSFIs conducted at LGS found station procedures to be adequate with some procedure discrepancies identified. The SSFIs conducted at PBAPS during the 1990 to 1992 time frame found a larger number of procedure discrepancies. With these identified procedure discrepancies considered, the SSFI teams did conclude that the operations, maintenance, and testing of the systems were adequate. A subsequent SSFI conducted on the PBAPS Feedwater System, in June 1993, after the completion of the major procedure re-write programs, found the procedures to be well prepared and a strength.

The SSFIs included specific tasks to assess the implementation of design bases in station operating, maintenance, and testing procedures. Each SSFI inspection team concluded that the selected system was capable of performing its safety function under postulated design basis accident conditions. Procedural discrepancies were identified and dispositioned.

b) UFSAR Verification Effort:

From May 1996 to September 1996, in response to industry activity involving UFSAR accuracy, PECO Nuclear performed a review of selected sections of the Peach Bottom and Limerick UFSARs. The purpose of the review was to assess the accuracy between the UFSAR and the facility. The Review Teams, which consisted of PECO Nuclear system managers and design engineers, reviewed the selected UFSAR sections for accuracy using Technical Specifications, Design Baseline Documents, referenced calculations, and other design bases information. Thirty (30) sections from each station's UFSAR were selected for review. Selection was based on Probabilistic Safety Analysis (PSA) significance and engineering judgment. During the course of the review, additional portions of the UFSAR were reviewed as a result of the internal cross referencing present in the UFSARs. UFSAR Section reviews performed by the Review Teams were reviewed by a Management Review Panel. The Management Review Panel consisted of engineering, operations, and licensing personnel.

Station procedures were reviewed against the UFSAR when these procedures provided the basis for assuring the accuracy of the UFSAR content.

The UFSAR Verification Effort identified about 450 discrepancies at Limerick and about 600 at Peach Bottom. Of these discrepancies, a limited number (<10) at each station were found between the UFSAR and station procedures which required revision to station procedures. In one of these instances at each station the significance of the issue warranted generation of a nonconformance report (NCR). The affected procedures are being revised to address the identified discrepancies.

The primary focus of the UFSAR verification effort was to self-assess the accuracy of the LGS and PBAPS UFSARs. This provided an opportunity to identify discrepancies between the station procedures and the UFSAR. A limited number (< 10) of discrepancies at each station were found between the UFSAR and the station procedures which required revision to the station procedures. No Unreviewed Safety Questions were identified.

C. NRC Oversight:

The following NRC inspection reports and associated corrective actions provide insights into the consistency between station procedures and the design bases. In those cases where the NRC inspection report cited specific procedure deficiencies, the subsequent corrective actions provided procedure and programmatic enhancements.

1. Limerick:

a) The NRC Electrical Distribution System Functional Inspection (92-81/81)(LGS)

The NRC conducted an Electrical Distribution System Functional Inspection (EDSFI) of LGS in 1992 to determine if the Electrical Distribution System (EDS) was capable of performing its intended safety functions, as designed, installed, and configured. The NRC inspection team concluded based upon the sample of design documents reviewed and equipment inspected, and taking into consideration the compensatory actions regarding the electrical bus transfers, that the Electrical Distribution Systems at Limerick are capable of performing their intended functions. Two violations with respect to station procedures and the design bases were identified.

- Additional Class 1E batteries loads shown on design calculations were not incorporated into the surveillance procedures, Technical Specifications, and UFSAR.
- The design requirement for a minimum battery voltage of 108Vdc stated in a design calculation was not incorporated into the Technical Specification Surveillance Test procedures.

The Root Cause Analysis performed by PECO Nuclear on these issues identified that the procedure for control of calculations did not adequately address the impact of changes to calculations on the UFSAR and Technical Specifications, and any associated Surveillance Test procedures. The calculation control procedure now requires calculation revisions to be processed utilizing the Engineering Change Request (ECR) process which requires performance of a 10CFR50.59 Review (if applicable) and a review for the impact of the calculation revision on station procedures.

As a result of the two issues identified by the EDSFI, the calculation control procedure was upgraded to require preparation of a 10CFR50.59 Review and a review of station procedures when performing a calculation revision. This provides an improved process for assuring calculation design bases information is correctly translated into station procedures and for identifying any impact on the Tecnnical Specifications and the UFSAR.

2. Peach Bottom:

a) NRC Safety System Functional Inspection (SSFI) (90-200/200)(PBAPS)

NRC SSFI on the PBAPS HPCI and ESW systems, (February-March, 1990) resulted in the following two findings related specifically to the control and use of design bases information:

"The inspection team concluded that design bases information for the ESW system was not adequately controlled and was not adequately supported by sufficiently detailed analyses or by appropriate test verification."

"With respect to the HPCI system, the team concluded that management attention was needed to correct the design, modification control, and maintenance deficiencies that were identified during its evaluation..."

Among the root causes that PECO Nuclear identified in its response to the SSFI Inspection Report was "there was a lack of understanding of the ESW design bases and changing system design requirements."

The 1990 NRC SSFI was a catalyst that caused PECO Nuclear to initiate several important program improvements. PECO Nuclear pointed to the following initiatives as barriers to prevent recurrence of the conditions identified by the SSFI:

- design baseline document efforts (DBD Project)
- improved 10CFR50.59 Review process
- improved modification process
- plans for future utility-conducted SSFIs.

In a follow-up report to the SSFI (Inspection Report 90-80/80), the NRC noted "the review indicated that program weaknesses had existed, but licensee efforts during the last two years have corrected many. A steady improving trend in these areas was noted by the team." The NRC further commented, "Perhaps the most noteworthy effort (to prevent recurrence) is the licensee's self-initiated program of internal SSFIs. This effort, in conjunction with the developing DBD program, should help to identify and address similar design, maintenance and test issues."

b) NRC Inspection Report 90-15/15 (PBAPS)

The NRC identified several discrepancies with PBAPS Maintenance procedures that did not contain quantitative or qualitative criteria to assure that procedural activities were completed satisfactorily. This issue of a general lack of acceptance criteria in Maintenance procedures was an NRC Inspector Unresolved Item. Corrective actions to address this issue included the implementation of the PBAPS Maintenance Procedure Re-Write Project described above that provided in the procedures written instructions to ensure data comparison and acceptance criteria sections when required.

Closure of this Unresolved Item (see Inspection Report 92-04/04) provided an independent assessment of the programmatic controls over Maintenance procedure format and content, and a review of a sample of individual procedures written in compliance with these controls. It was concluded that the Maintenance procedures were well written and contained adequate information.

c) NRC Inspection Report 90-17/17 (PBAPS)

In September 1990, a review of High Energy Line Break (HELB) controls at PBAPS was initiated because of concerns identified at LGS. It was determined that the program to control penetrations did not address controls for HELB barriers. The corrective actions for this issue included walkdowns, resolution of non-conforming conditions, and the development of new design and procedural controls for barriers.

Design controls and installation and maintenance processes were established to provide the appropriate controls for breaching HELB barriers, including the requirement to perform 10CFR50.59 Reviews. The inspection report unresolved item was closed with the conclusion that the efforts to resolve the HELB issue were very thorough.

d) NRC Electrical Distribution System Functional Inspection (EDSFI) 93-80/80 (PBAPS).

The inspection report noted, "the team reviewed operating procedures (normal, abnormal and special event) to confirm that the operating instructions and administrative controls were adequate to ensure operability of the EDS under all plant operating conditions. Based on the review performed, the team concluded that the procedures contained a sufficient level of detail to ensure that the procedure objectives could be accomplished satisfactorily even though there were some minor weaknesses."

The scope of the inspection included verification that critical design parameters were adequately incorporated into procedures. There were no findings regarding weaknesses in incorporating design bases information in procedures, although there were no explicit statements in the inspection report addressing the adequacy of linkage between design bases information and station procedures.

e) NRC Inspection Report 95-27/27 (PBAPS)

NRC inspection determined that operating procedures for the Standby Gas Treatment System (SBGT) allowed the system to be run in modes that could have prevented the system from performing its design bases function if a single failure were to occur during a design bases event. This condition resulted in an NRC Notice of Violation (NOV 96-06-03).

PECO Nuclear immediately corrected the problem by revising the System Operating Procedures (SOs) such that the design bases function of the SBGT system would not be lost during routine system operation. PORC opened an item to review other system operations to determine if other system operating procedures allowed modes of operation which could prevent a design function.

PECO Nuclear's investigation of this event is documented in a PEP Issue. It was determined that the operating procedure in question was originally drafted in the late-1970s and that the plant's licensing bases were not adequately reviewed when preparing the Standby Gas Treatment (SBGT) system operating procedure. It was noted that with current administrative controls and a SBGT DBD in place, present day procedural and design guidance is vastly improved. The assessment of generic implications from this issue included whether other operating procedures allow system alignments not described in the UFSAR, and whether procedures preserve the affected system's ability to withstand a single failure when required by design basis assumptions. The review of procedures for system alignments not described in the UFSAR was completed in conjunction with the UFSAR Verification Effort. Several other systems could be aligned in a manner not described in the UFSAR. These issues were entered into the appropriate corrective action process and resolved. The review to determine whether procedures preserve the affected system's ability to withstand a single failure is in progress and is being tracked by the PEP process.

D. Conclusion:

When considered in total, the various rewrite projects resulted in an upgrade to many maintenance, operating, and testing procedures. No single project was explicitly required to verify every aspect of the design bases; however, every project was required to verify its procedures against one or more aspects of the plant's design bases.

Corrective actions associated with NRC oversight activities have resulted in improvements to programs and processes which are critical to maintaining the accuracy of design bases information in station procedures. The areas which were upgraded include: the engineering change/modification control process, Instrumentation and Control (I&C) configuration control, the Environmental Qualification process, engineering calculation control, the commitment annotation process, the 10CFR50.59 Review process, and design baseline document development.

Internal PECO Nuclear SSFIs (4 at LGS, 6 at PBAPS) were performed in substantial technical depth and specifically assessed implementation of the design bases in operations, maintenance, and testing. The SSFIs at both stations identified some procedure discrepancies, but all systems which were inspected were determined to be capable of performing their design bases functions.

The remaining external and internal assessments (design baseline document development, UFSAR verification, NQA audits) were generally less rigorous than the SSFIs in addressing the linkage between design bases information and maintenance, operating, and testing procedures. They were; however, much broader in terms of the number of systems and procedures assessed, and provide an indication of overall procedure health. These assessments found a number of minor discrepancies between station procedures and design bases documentation; however, no safety-significant issues were identified.

Therefore, when the various rewrite projects are considered in concert with the outcome of the internal and external assessment activities, reasonable assurance exists that current maintenance, operating, and testing procedures adequately reflect the plant's design bases.

II. PROCEDURE CONTROL:

A. Procedure Control Process

All procedures which contain plant design bases information utilize the same revision process, the Station Qualified Reviewer (SQR) process (A-C-4.2). The procedure Temporary Change process (A-3) also utilizes the SQR approval process. There are approximately 18,000 procedures total for PBAPS and LGS. This process, or one with equivalent levels of review and approval, has been in use at both PBAPS and LGS since 1989.

Within the normal process (A-C-4.2), the preparer of a new or revised procedure:

- prepares the document in accordance with AA-C-5, "Preparation and Control of Procedures/Guidelines," A-C-1, Procedure Writers Guideline, and AG-CG-91, Processing

ATTACHMENT 1

Procedural Documents and Revisions, which provide direction for technical reviews, human factoring, formatting/style and standard writing practices

- verifies that scope and content is sufficient to implement source document requirements. (Reg. Guides, UFSAR, Tech. Spec., QA Program LGS UFSAR CH.17/PBAPS UFSAR App. D, Policies and Directives, Vendor technical manuals, Design Baseline Documents and OEAP items)
- ensures quantitative and qualitative acceptance criteria, sign-offs, verification points per A-C-33, "Process for Verification of Quality," are included to assure important activities are satisfactorily accomplished
- evaluates the impact on interfacing procedures or interfacing plant systems per AA-C-5
- takes action to have interfacing procedures revised
- initiates appropriate training
- reviews for licensing commitments to ensure they are contained within the procedure per LR-C-1, Commitment Tracking Program
- reviews for recommended enhancements per AG-CG-87, Procedure Performance Improvement System
- performs or obtains a 10CFR50.59 Review per LR-C-13, "10CFR50.59 Reviews," to determine whether the change involves 1) a Technical Specification change or other Facility Operating License amendment or 2) a change to the facility or procedures as described in the SAR or 3) a test or experiment not described in the SAR. If the procedure involves any of these, a Safety Evaluation is performed to determine whether the procedure involves an Unreviewed Safety Question

The SQR:

- is appointed by the Plant Operations Review Committee (PORC) Chairman as a knowledgeable person in their functional area
- performs an independent review of new or revised procedures for technical accuracy, conformance to station procedures/programs/policies and commitments
- reviews the associated 10CFR50.59 Review
- ensures that a cross-disciplinary review is performed, as necessary
- recommends approval of acceptable procedures to the Responsible Superintendent (RS)

The Quality Reviewer (QR):

- is appointed by the PORC Chairman or Chesterbrook Program and Procedure Section Manager, as a knowledgeable person in quality requirements in their functional area
- reviews administrative procedures pertaining to quality related activities to assure quality

requirements are met or maintained in the procedures. This includes Federal Regulations, UFSAR, Tech Spec, QA Program Description, Commitments, Policies, and Directives

The Responsible Superintendent:

- is appointed by the PORC chairman as a knowledgeable person in their functional area
- reviews new or revised procedures for the area which he/she is responsible for nuclear safety issues and conformance to station procedures, policies, and commitments
- concurs with the determination of whether or not a 10CFR50.59 Safety Evaluation is required. If an Unreviewed Safety Question is involved then appropriate NRC approval is obtained prior to implementing the activity
- approves acceptable new or revised procedures that do not require PORC review

If the activity involves:

Administrative Procedure as defined in the Technical Specifications,
Procedure/Guideline requiring a 10CFR50.59 Safety Evaluation,
A Special Procedure (LGS only),
Emergency and certain Security Plan implementing Procedures,
Other procedures as directed by management as requiring PORC,
Off-site Dose Calculation Manual,
Radwaste Process control plan,

then PORC review and the Plant Manager's approval are required.

Within the Temporary Change (TC) Process (A-3), the preparer of a revised procedure:

- determines that the TC clearly does not change the intent of the procedure
- ensures the TC does not inappropriately change commitments
- requests approval for TC implementation from an SQR knowledgeable in the area affected and a Senior Licensed Operator

The SQR and Senior Licensed Operator:

- verify the intent of the procedure is not changed
- provide independent review for technical accuracy
- designate an Independent Technical reviewer if required based on the complexity and perceived need for such review

Additional SQR and RS approvals are required within 14 days per the Technical Specifications.

B. Evaluation of Procedure Control Process

Documents governing procedure revision activities provide direction to consider design basis information when creating or revising procedures.

The following assessment of a conversion of Logic System Functional (LSF) Tests supports the adequacy of the procedure revision process in maintaining design basis information in procedures.

In 1995, a performance, compliance, and technical based NQA assessment was conducted at Limerick to verify the adequacy and effective implementation of the administrative controls governing the Surveillance Testing and Logic System Functional Testing programs. The Assessment Team reviewed six Action Requests associated with ten Technical Specification Amendments, and three STs which were impacted, verifying that identification of STs requiring revision, and the tracking of these revisions until completion was performed. Additionally, the Team performed a technical review of eleven High Pressure Coolant Injection LSF STs against P&IDs and logic prints to verify that the STs addressed the testing of all the logic associated with the HPCI pump suction valves from the Suppression Pool and the Condensate Storage Tank. NQA concluded that the Administrative Controls for the ST and LSF Testing Programs are adequate and being effectively implemented in a manner which ensures conformance to LGS Technical Specifications and regulatory requirements.

A review of PEP issues indicated that some deficiencies have occurred in station procedures due to inadequate review and/or incorporation of design information. Examples include:

1. Steps were added to the LGS RCIC Pump, Valve and Flow surveillance test to dial the RCIC Area Radiation Monitor (ARM) down to zero so that it would not alarm during the RCIC run so that entry into TRIP procedure T-103 ("Secondary Containment Control") would be avoided. UFSAR Section 12.3.4.1.1.B, however, states that the design bases of the ARMs is to provide a record and continuous indication in the Control Room of gamma radiation levels at selected locations within the various plant structures. Peer review of the 50.59 review for the procedure revision did not consider the design bases of the Radiation Monitoring System; only RCIC's design bases were considered. Corrective action included training to sensitize 50.59 preparers & reviewers to the need to consider all systems affected by the proposed activity.

2. Several PBAPS Surveillance Test valve stroke time acceptance criteria were changed without considering stroke time limits contained in the UFSAR. The new acceptance criteria were less conservative than the UFSAR values. The two root causes for this event were: (1) failure to recognize that valve stroke time acceptance criteria are based, in part, on UFSAR limits, and (2) the relevant UFSAR section is not referenced in appropriate test procedures. Corrective action included: (1) a comprehensive review of all valve stroke time acceptance criteria, and (2) revising the IST Program document to include a reference to the relevant UFSAR section. This condition was identified during the Improved Tech Spec (ITS) implementation program.

A review of the procedure preparation process indicates that mechanisms are in place which require the preparer to consider the full spectrum of design bases requirements. Documents governing procedure revision activities provide direction for consideration of design bases information when creating or revising procedures. The Temporary Change process uses a screening process and approval process that provides barriers to ensure design bases information is not changed. The review process (SQR, RS, PORC) addresses important aspects of the design bases (e.g., Technical Specification requirements, regulatory commitments). A review of PEP issues associated with deficient procedure control process activities identified no generic process weaknesses. PECO

ATTACHMENT 1

Nuclear concludes that there is reasonable assurance procedure revision program requirements are being implemented adequately.

III. CONTROL OF DESIGN BASES:

As discussed in response to Request (a), the Engineering Change Request (ECR) process as delineated by procedure MOD-C-9 is used to control, process and document most changes to design and licensing bases as directed by governing procedures. The processes which control design and licensing bases changes have provisions to maintain operating, maintenance, and testing procedures current.

MOD-C-9 directs the person processing the ECR to:

- determine the impact of the design and/or licensing bases change on station procedures and programs
- use the Station Document Checklist in the MOD-CM-1, the Modification Manual, as a guide which delineates procedure and program types and the organizations responsible for implementing the changes. MOD-CM-1 contains an individual Station Document Checklist for LGS and PBAPS.
- initiate tracking documents in PIMS to request assistance by the appropriate station personnel in evaluating the impact on station procedures and programs. MOD-C-9 also provides direction to use MOD-CM-1 which provides direction on the method to revise and track to completion the revisions.

For those processes that allow changes outside of the MOD-C-9 process (e.g., specifications, fuel management, and certain software changes) the processes provide procedural direction to identify, evaluate, track, and process procedure changes.

A. Evaluation of Design Bases Control Processes:

An NQA joint assessment was conducted at Chesterbrook, Limerick, and Peach Bottom Generating Stations in 1994 to evaluate the adequacy and effective implementation of the Nonconformance Report (NCR) Process. Review of forty-five (45) NCRs indicated that required procedure changes were completed and correct. NQA concluded that changes as a result of NCRs were properly evaluated, posted, and effectively tracked.

As part of the corrective actions for an open Safety Relief Valve (SRV) event in September 1995 at Limerick, Station Engineering evaluated Instrument Setpoint Changes Requests (ISCRs) to assess the potential to mask equipment or system performance problems. The Independent Safety Engineering Group (ISEG) independently evaluated thirty-five (35) ISCRs. ISEG performed this assessment by reviewing the Engineering dispositions of ISCR type ECRs, UFSAR, Technical Specifications, procedures, items in PIMS, Instrument Index, and selected Troubleshooting Control Forms (TCFs). ISEG concluded that the setpoint changes via ISCRs were appropriately controlled.

A review of the processes for changing the design bases indicates that the processes have the procedural mechanisms to identify the impacted station procedures and revise them. However, numerous assessments have indicated that implementation of the process requires strengthening to assure all impacted procedures are updated as required.

A review of NRC Inspection Reports, PEP issues, NQA audits, Self Assessments, and ISEG Reports indicates a number of deficiencies have occurred in station procedures due to the inadequate incorporation of design changes into the procedures. Recent data has indicated that the corrective actions taken have not adequately resolved this problem.

Procedure Revision Cause Code Data is generated from assessments made during the procedure revision process to identify the reason for the revision and whether a process was followed appropriately or there was a process deficiency that necessitated the revision. This analysis tool was developed in part as a means to monitor the effectiveness of the processes which identify and implement revisions to procedures.

In 1996, there were approximately 10,000 procedure revisions performed. Fifteen percent (15%) of these revisions were due to implementation of design changes. Of these revisions required by design change implementation, 8.8% were made after the implementation of change (1.3% of all revisions), and; therefore, represent a weakness in initially identifying affected procedures. To analyze the data, each revision that makes up the 1.3% was reviewed to determine whether a potentially significant operational impact resulted from failing to initially identify the affected procedure. The determination of potentially significant operational impact was made using the collective judgment of a team of experienced engineers and professionals to assess whether the missed revision could have caused equipment not to function properly, personnel to take an inappropriate action, or the response to an event to be inappropriate. The review concluded that there was one procedure that was of concern which was previously reported to the NRC in Limerick LER 1-96-015. As analyzed in this LER, the actual and potential safety significance of this missed procedure revision was minimal.

At both stations, the expectations surrounding procedure use are delineated in a common administrative procedure titled, "Procedure Use and Adherence." These expectations, along with supervisory coaching, have led to a work force that is intent on identifying and resolving discrepancies with procedures. This is reflected in the number of procedure revisions that occur each year.

A review of the processes for changing the design bases indicates that the processes have the procedural mechanisms to identify the impacted procedures and revise them. However, the assessments indicate that, although the procedural mechanisms are being performed, the implementation of the process requires strengthening to assure all impacted procedures are consistently updated as required. The number of procedures that require revisions due to design changes is small compared to the body of procedures that implement design bases requirements and the processes for updating procedures due to design changes are generally being followed. A review of the 1996 procedure revision cause code data found that the missed procedure revisions had minimal impact on plant safety. Therefore, there is reasonable assurance that procedures adequately reflect the station design bases. However, an assessment of the recognized weakness to identify and revise procedures impacted by design bases changes will be performed and appropriate corrective actions implemented (see commitment 4 in Attachment 3).

IV. SUMMARY CONCLUSION:

Considering the projects, assessments, and corrective actions discussed previously, it is concluded that there is reasonable assurance that the current PBAPS and LGS maintenance, operating, and testing procedures adequately reflect the station design bases.

A review of the procedure control process indicates the process appropriately addresses design bases requirements and has been adequately implemented.

A review of the processes for changing the design and design bases indicates that the processes have the procedural mechanisms to identify the impacted station procedures and revise them. However, the assessments indicate that, although the procedural mechanisms are being executed, the implementation of the process requires strengthening to assure all impacted procedures are consistently updated as required. The number of procedures that require revisions due to design changes is small compared to the body of procedures that implement design bases requirements and the processes for updating procedures due to design changes are generally being followed. A review of the 1996 procedure revision cause code data found that the missed procedure revisions had minimal impact on plant safety. Therefore, there is reasonable assurance that procedures adequately reflect the station design bases.

ATTACHMENT 1

REQUEST (c)

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

RESPONSE

Throughout the course of plant operations, the plant's configuration and performance would be expected to change as the licensee has intentionally modified the plant for regulatory and performance reasons. Regulations require that reviews and updates of the plant design and licensing bases be performed when making changes to plant structures, systems and components (SSC). Because the SSCs are not all necessarily products of current standards, rationale for concluding that the configuration and performance of the SSCs are consistent with the design bases must consider the historical aspects and rely on comprehensive assessments that have occurred. A review was performed which indicates that Peach Bottom Atomic Power Station's (PBAPS) and Limerick Generating Station's (LGS) configuration and performance are consistent with the design bases.

The review focused on those programs, inspections, and assessments that have occurred during plant life and provided opportunities to measure consistency between the design and the hardware. Each program reviewed was evaluated for its comprehensiveness and quality; findings and corrective actions were also considered as were the programs' time frames to ensure that value credited is consistent with the standards and expectations of the time.

Periodic and special testing monitor key aspects of system and component performance which demonstrate that systems and components remain capable of performing their intended functions. This testing includes surveillance testing, post-maintenance testing, and modification acceptance testing. Test acceptance criteria are derived from design basis requirements. Systems or components failing to meet test acceptance criteria are evaluated and restored using appropriate corrective action processes. These activities are continuous over plant life and ensure that plant system and component performance meets acceptance criteria, and also provide reasonable assurance that system and component performance is consistent with the design basis.

I. PROGRAMS INITIATED TO VERIFY SSC RELATIVE TC THE DESIGN BASES:

Several programs whose primary purpose was to measure SSC configuration and/or performance relative to the design bases were identified by the review. These programs provide reasonable assurance that SSC configuration and performance are consistent with the design bases because they involved identification of specific design versus plant issues and analysis of the data's relevance to PECO Nuclear's ability to assure all SSCs remain consistent with the design bases. Unless stated otherwise, the programs described were/are common to both stations.

A. Safety System Functional Inspection (SSFI) Inspection Results

SSFIs were conducted on a sample of systems from 1990 to 1994. The objectives of the SSFIs were to determine if the systems as designed, installed and configured were capable of performing their intended safety functions and to determine the effectiveness of engineering and technical support processes as they relate to the systems' ability to perform their intended safety functions. SSFIs were conducted by PECO Nuclear at LGS on High Pressure Coolant Injection, Feedwater, Emergency Service Water, and Diesel Generators and AC Electrical Distribution systems. SSFIs were conducted by PECO Nuclear at PBAPS on 125/250 VDC Distribution, High Pressure Service

ATTACHMENT 1

Water, Diesel Generator and AC Emergency Electrical Distribution, Stand-by Gas Treatment, Compressed Air and Feedwater Systems. SSFIs were conducted by the NRC at PBAPS on the Emergency Service Water System and High Pressure Coolant Injection System and on both the LGS and PBAPS Electrical Distribution Systems. The SSFIs employed a "deep vertical slice" methodology and team interaction techniques developed as detailed in NSAC 121, November 1988. Walk downs and technical reviews of studies, calculations, design drawings, modifications, surveillance tests and corrective actions for previously identified deficiencies were performed. PECO Nuclear and NRC SSFIs were similar in their methodology and depth of review, and they arrived at similar conclusions, with the exception of the Peach Bottom Emergency Service Water System which is discussed in paragraph 2 below.

1. LGS Results

All SSFIs concluded that the systems were capable of performing their intended functions under postulated design bases accident conditions. No operability or significant safety concerns existed. Systems were confirmed to be adequately sized and configured and to have sufficient margin to ensure acceptable performance over time. Inconsistencies identified between calculations and testing procedures were bounded by design margins. Engineering/Technical Support organizations were found to be staffed with competent personnel. Good practices were evident in root cause analyses, modification packages, non-conformance report (NCR) evaluations and 10CFR50.59 Reviews. Maintenance and testing programs were acceptable and required parameters were being adequately tested. Generally, staff and operator knowledge and training programs were strong.

The Limerick NRC EDSFI resulted in two violations pertaining to the incorporation of loads in battery calculations and incorporation of battery calculation results into Technical Specifications and surveillance tests. These violations were severity level IV as combined with a third violation concerning a human performance incident having no bearing on system design or system performance. Surveillance tests were revised to ensure complete acceptance criteria were specified and the load cycle tables have been transferred from the Technical Specifications to the UFSAR where they are maintained.

2. PBAPS Results

In 1990 at PBAPS, the SSFI on the ESW system concluded that the design bases information for the ESW system was not adequately controlled and was not adequately supported by sufficiently detailed analyses or by appropriate test verification. The identified issues with ESW were resolved and since that time SSFIs have shown in each case that the plant design bases are consistent with plant configuration and performance. The initiatives and programs which have been underway or completed since the ESW SSFI have resulted in much improved control of plant configuration and design bases. This is supported by the findings in later SSFIs.

While the SSFIs have shown consistent strengths in material conditions they have also identified the need to better document the design bases, correct minor errors in the calculations and generally pay more attention to design bases. In several cases, margin was used unknowingly or the design bases were not documented effectively. With the exception of ESW, none of the errors was significant enough to cause a failure to meet the design bases.

3. LGS and PBAPS Conclusions

Control of calculations needed improvement. Many instances of duplicate calculations, calculations that had been superseded but were not identified as such, inconsistent overlapping calculations, calculations that had not been updated to reflect current configuration or conditions, and calculations which could not readily be determined to be consistent with plant configuration or testing procedures were discovered. Also, the lack of an integrated calculation index made it difficult for engineers to find calculations of interest. This problem has been initially addressed by having the Design Baseline Documents (DBD) reference calculations for "controlling parameters." Additionally, significant improvements were made in the design control processes relative to calculations. Procedures require that calculations be reviewed and updated when plant changes are designed, that overlapping calculations (existing and new) be annotated as such when plant changes are made, and that annotations be made to identify inputs to and outputs from other calculations when plant changes are made. Furthermore, these measures assure that calculation control and maintenance will support the design basis.

Performing SSFIs on a sample of critical systems is an accepted and sound approach to validating the design and implementation processes common to all safety related systems. The sample systems were selected based on their high significance in plant licensing as well as their high Probabilistic Safety Assessment (PSA) significance in preventing or mitigating design bases events. Thus, an independent and objective assessment of systems and processes critical to assuring the public health and safety was accomplished by the SSFIs. With the exception of the ESW system at PBAPS, no significant process flaws were indicated by the SSFI results. Corrective actions were implemented, as appropriate, and tracked to completion for each issue raised by the SSFI teams. The overall results of the SSFI program and subsequent improvements to the design control processes support the conclusion that the plant configuration and performance are now and will continue to be maintained consistent with the design bases. A review of the SSFI results in aggregate has identified a potential concern relative to past calculation control and the potential impact on the current plant configurations/design bases. A PEP issue has been generated to investigate this potential concern and to implement necessary corrective actions.

B. UFSAR Verification Program

The purpose of the UFSAR Verification effort was to develop and implement a methodology for verifying and remedying the accuracy of selected information contained in the Limerick and Peach Bottom UFSARs. The review of the UFSAR documents was formally implemented, at both Stations, from May through September 1996. This review program was a proactive response by PECO Nuclear Senior Management to recent industry events.

The scope of the UFSAR verification program included an original scope of 30 UFSAR sections for each station. The original selection criteria for sections reviewed included PSA significant items, sections with high change activity and random sections. The scope was expanded during the review process, at the direction of the section review leads, to include an additional 53 sections at PBAPS and an additional 68 sections at LGS, resulting in a total review of approximately 20% of each UFSAR. The total review process and corrective action follow-up involved an estimated 4 man-years per UFSAR reviewed. The program scope included identification and resolution of identified inconsistencies with the document and the as-built/as-found facility.

The program involved the following review activities: (1) each team was assigned a section to review (each section was selected as describing a plant system), (2) section review expanded, in most cases, to encompass additional sections referenced from within the original section, (3) inconsistencies were categorized as "typos," "incorrect statements" and/or "ambiguous statements" and assigned a significance level based upon safety and/or regulatory impact, (4) management review panel for concurrence of review depth and proper characterization of any findings, and (5) review by the PECO Nuclear Engineering Council. The management review panel consisted of station management, engineering, operations, and licensing personnel.

ATTACHMENT 1

The review identified about 450 discrepancies at Limerick and about 600 at Peach Bottom. The largest percentage of these discrepancies included statements within the UFSAR which were categorized as "ambiguous", i.e., information contained within the UFSAR which could be misinterpreted. "Incorrect" statements accounted for approximately 9% (at Peach Bottom) and 16% (at Limerick) of the identified discrepancies. However, a small number of incorrect discrepancies (5 at LGS, 6 at PBAPS) were determined to be nonconforming conditions. No incorrect statements identified resulted in an Unreviewed Safety Question (USQ).

Recommendations for process enhancements were provided to the Nuclear Engineering Council following the program team's initial assessment. Recommendations included implementing a guideline for administrative control and maintenance of the Personal Librarian Software (PLS) database and overall changes to the ownership of the UFSAR sections. The use of the PLS tool, which facilitates SAR reviews for 10CFR50.59 Reviews, must be communicated and demonstrated to all personnel involved with the 10CFR50.59 Review. Training for organizations outside of the engineering organization and to new engineering personnel must be focused to include sensitization to the information contained within the UFSAR and overall UFSAR fidelity. This training would be incorporated into station continuing training lesson plans. A PEP issue was generated to evaluate and identify any Conditions Adverse to Quality (CAQ), root causes, generic implications and corrective actions through a root cause analysis. The PEP process will also track the previously listed recommendations to closure.

Due to the volume of discrepancies identified, a few remain tracked as open items within the PECO Nuclear corrective action system. All nonconforming conditions identified during this review have been dispositioned via common procedures A-C-901 and MOD-C-9. All other discrepancies identified, with regard to the "incorrect", "ambiguous", or "typos" are being evaluated and resolved, in an expedited manner, via the Engineering Change Request process (parent procedure MOD-C-9). Changes to the UFSAR are undergoing a review per the criteria in common procedure LR-C-13 and a 10CFR50.59 Review, as applicable.

PECO Nuclear is committed to complete the verification of the UFSAR and is evaluating the scope and schedule of the completion of the UFSAR verification and will provide additional information in future correspondence (see commitment 1 in Attachment 3). Closure of identified discrepancies continues. Corrective actions associated with the program concerns will be identified and tracked via the PEP process once the review is completed. Additional baseline and continuing training of nuclear personnel will provide continued focus on UFSAR content, accuracy and fidelity.

C. Power Rerate Program (LGS and PBAPS)

The Power Rerate Program scope included all necessary activities to assure safe, reliable operation of the plants at 105% power and included a comprehensive review of plant design bases. Both systems operation and current design bases were thoroughly reviewed for impact of 105% power operations. Prior to the start of this review, PECO Nuclear Engineering obtained input information on current system performance data from interviews with System Managers. Calculations were reviewed and revised as necessary. System Review Boards/PCRC Reviews were used to validate Engineering's assumptions and results of their reviews. A comprehensive testing program was implemented to assure expected results at 105% power. The rerate program began in 1992 with final acceptance testing at rerated conditions completed during Unit refueling outages in 1994, 1995, and 1996.

Power Rerate involved a comprehensive review of plant design bases to determine if the existing controlling parameters affected by Power Rerate were consistent with the plant and able to remain within design bases at 105% power. Extensive System Manager involvement in the program assured that the engineering data, including the rerate inputs and performance data and the calculation outputs matched their knowledge of the system operation and system configuration. System performance data was used to assure degradation of equipment over plant life and replacement equipment was incorporated into the analysis. Comprehensive design review and recalculation of calculations was performed to assure that the units would operate per design at 105% power. In the vast majority of areas, the plant and design bases were consistent and calculations were either recalculated at 105% power conditions or reviewed to determine acceptability at 105% power conditions. Where the design was not consistent or could not be verified, the calculations were recalculated or regenerated.

Significant startup testing was performed at each of the units following implementation of Power Rerate. A structured review of rerate results was performed to assure that the results were consistent with the System Managers' and plant management's understanding of current plant configuration. This comprehensive testing program assured plant performance met expectations.

This comprehensive design review, regeneration, and testing program provides strong support of the rationale that the plant configuration and performance is consistent with the design bases.

D. Component Record List (CRL) Development

The CRL originated as a configuration management effort to consolidate component information which, at the time, was contained in several different lists and databases such as the Q List, fuse index, and Environmental Qualification list. Following the initial data acquisition effort, the CRL has evolved to provide increasing functionality as needs were identified. Presently, the CRL defines component IDs for the plants, includes the Q List for PBAPS and LGS, Codes and Classifications information, Nuclear Plant Reliability Data System (NPRDS) data, Equipment Qualification information, and numerous other data on plant components and contains component design bases for the plant. Component labeling and nomenclature is almost entirely based on the information in the CRL; this referencing standard is used throughout PECO Nuclear processes.

Verification activities for the CRL primarily occurred during data loading of several previously existing data bases and did not normally include field verification activities. Field verification activities were typically performed during resolution of problems identified aside from the CRL data loading activities.

The CRL plays a critical role in plant design and operation. Routine activities, such as maintenance testing and procurement, interface with the CRL which makes it easier to identify discrepancies between CRL data and the physical plant. Discrepancies are resolved using appropriate corrective action processes. Each of these processes incorporates a "generic implication" review to determine if an identified issue is a random occurrence or an indicator of a bigger concern and may include field verification. Additionally, several internal self assessments have been performed on the CRL during its existence. Each focused on particular CRL data and identified and corrected discrepancies, further improving data quality. The general approach of corrective action processes and self assessments provides a continuous improvement mechanism to increase the confidence that data in the CRL is consistent with the physical plant.

E. Inservice Inspection / Testing Program Development

The requirements and scope of the Inservice Inspection (ISI) and Inservice Testing (IST) programs are defined in the Technical Specifications and UFSARs at both PECO Nuclear Stations. Station procedures and specifications exist to implement, control, track, and summarize the inspection and test programs. The baseline walkdowns and subsequent as-built reconciliation programs were completed as part of the pre-service program for the stations. Ongoing/continuing physical inspections and visual inspections are scheduled and will be completed throughout the life of the plants.

The ongoing program consists of 1) the physical walkdown and visual inspection of equipment included within the program, and 2) the performance testing of these components at defined frequencies. The testing and visual inspections of these components are completed to verify that the installed components continue to perform in accordance with the design requirements of the as-built system. During the inspection and testing, any discrepancies identified are documented and resolved by appropriate corrective action processes.

These programs and processes are assessed internally by both the NQA and Engineering Assurance organizations. A review of various assessment documents shows that overall, the programs are identified as being favorable. Over the previous two years, improvements have been completed. Improvements identified and implemented include reducing the number of components in the IST program and some of the requirements associated with the program. These changes were reviewed and documented under 10CFR50.59 Reviews. Revisions to implementing procedures have been completed to clarify process and material control requirements. Incorporation of the snubber programmatic and technical requirements into the ISI program was completed. A revision to procedure A-C-80 has been issued to streamline the procedural requirements, i.e., combination and deletion of ten procedures/guidelines.

The Inservice Inspection and Inservice Testing programs are reviewed by the NQA organization. Current self assessments are completed annually by the station engineering organizations. Open items being internally tracked within the corrective action process, include a review of NUREG-1482 guidance for impact at the stations, incorporation of the PECO Nuclear position developed for Generic Letters 90-05 and 91-18, replacement/ upgrading particular types of snubbers and revising the current information in the CRL to accommodate and accurately reflect IST component data. While there have been findings previously identified, documentation shows that the corrective actions were completed in a timely manner and the overall performance of the program has been good. The review process monitors the effectiveness and adequacy of these programs to maintain the design configuration of the systems, structures and components in the plants.

F. Generic Letter 89-10 Program

The Generic Letter 89-10 Program required PECO Nuclear to perform design bases review, documentation, regeneration of calculations, development of baseline performance data and trending of testing data for motor operated valves (MOV) in the program. The program developed improved processes for valve and valve operator maintenance and performance. The program was initiated in 1989, has been continuously assessed and improved, and is an on-going commitment.

MOVs are comprehensively monitored and their design information closely controlled. Plant walkdown data has been entered into a controlled computer database (MIDAS) and is updated on an ongoing basis. Plant design calculations were regenerated in the 1992 time frame to provide input into a controlled software program (MIDACALC) for valve settings.

MOV performance is routinely monitored and assured by ongoing testing and maintenance. Test results are input into the MIDAS database and failure trending and analysis is performed. Additionally, differential pressure testing is performed which corroborates MIDACALC software design settings and diagnostic evaluation.

Based on the regeneration of calculations, strong programmatic controls, accessible documentation of the current program, ongoing and prior NRC inspections, and ongoing performance testing, there is reasonable assurance that the plant design bases are consistent with plant configuration and performance for MOVs in the Generic Letter 89-10 scope.

G. Limerick Independent Design Verification Program

The review of Limerick's design was completed as part of the Plant Readiness Certification documentation presented to the NRC as part of PECO Nuclear's request for an operating license. The Unit 1 deep vertical slice review of the Core Spray System represented a sample of the design process employed by the Architect-Engineer (A/E). The final review of the report, by the NRC, was completed and documented in SER Supplement 3, Section 17.5. Unit 2 completed a detailed design review of the Containment Heat Removal mode of the Residual Heat Removal (RHR) system. This review, similar to the Unit 1 review, also utilized, in part, a physical walkdown approach to verify that the system was as-built to applicable and appropriate design criteria. The final review of the report, by the NRC, was completed and documented in SER Supplements 7, 8 and 9.

The two independent assessments completed for LGS identified discrepancies with the electrical loading and control of load changes. The corrective actions taken have been verified as still in place and controlled via common implementing procedures. The thermal overload calculation methodology needed to incorporate effects of low voltage and high ambient temperatures along with the negative tolerance associated with the heater design. This criteria is now included as part of the design criteria in a common procedure. A formalized explanation of how any existing design basis documents will be used when making modifications to the facility was required. This assessment was evaluated and completed during the design input document completion. The basis of the original design calculation for electrical load studies and short circuit studies needed to be re-assessed and a methodology for load control was to be established. The assessment was completed during the final calculation turnover to PECO Nuclear and the "load change evaluation form" is now completed utilizing a common procedure.

H. Fire Protection Programs

During 1994 the post-fire safe shutdown program manual was implemented; it documented guidance for program activities such as periodic Safe Shut Down (SSD) database reconciliation, Integrated Nuclear Data Management System (INDMS) (cable management/Appendix R analysis software) update requirements, document control requirements, component record list SSD guidance, and guidance on performing plant change evaluations. Also during 1994 the Individual Plant Examination of External Events (IPEEE) fire risk analysis and Thermo-Lag reduction projects were initiated. The Thermo-Lag reduction project was established to support resolution of the open issue, Generic Letter 92-08, the operability of the fire barrier material Thermo-Lag 330-1. Generic Letter 92-08 identified that protection provided by fire barriers constructed of Thermo-Lag 330-1 material manufactured by Thermal Science Inc. was indeterminate. The Thermo-Lag Reduction Project was chartered with resolving the issue. The resolution strategy for both LGS and PBAPS is to revise the safe shutdown analyses to reduce the reliance on the use of fire barriers (raceway encapsulations), then bring the required barriers into full regulatory compliance. This effort, which includes re-documenting some fire protection issues and the SSD analysis, is ongoing. Verification of SSD data was performed by documentation review when possible. However, much of the data was verified by field walkdowns. In addition to the re-analysis effort, certain issues associated with Thermo-Lag fire barrier material require the review of existing calculations The issue of Thermo-Lag combustibility required that the steel survivability and plant combustible loading analyses be revised to account for the quantities of the material in the plant. To revise the combustible loading analysis, the supporting calculation was automated using the cable management system (INDMS).

In recent years, numerous audits (short duration reviews targeting specific issues) of the fire protection program bases have been performed without significant findings identified. This includes audits performed by the NRC, ANI, and PECO Nuclear QA organizations. However, the Thermo-Lag Reduction Project, a detailed verification of issues/data associated with revising the safe shutdown analyses, has self-identified several safe shutdown nonconformances. The Thermo-Lag Reduction Project is tracking and evaluating nonconformances identified in an attempt to identify generic issues. None have been found. The most recent nonconformance identified was for the LGS remote (alternative) shutdown method. An assumption used in a technical justification was identified as being non-conservative. The nonconformance was resolved by a minor plant modification and rework of a proceduralized safe shutdown repair. A PEP issue is in progress to evaluate the nonconformance.

Recent LERs concerning implementation of the fire protection program at LGS have been identified and promptly corrected. "Lessons Learned" reviews performed for each issue have identified no similar issues at PBAPS. The most recent issue was regarding unit startup with safe shutdown required valves not being administratively controlled. A PEP issue has been initiated to evaluate the event.

i. Station Procedure Rewrite/Review

Different levels of procedure rewrite/review programs were performed at Peach Bottom and Limerick stations. Each station's effort is discussed separately.

1. Limerick

LGS System Operating, Off-Normal, Operational Transient, Special Event, Event, and Transient Response Implementation Plan, Operations Department Surveillance Test and Operations Department Routine Test Procedures were reviewed as part of a comprehensive procedure upgrade project between August 1993 and December 1995.

Procedures were verified to be consistent with the plant configuration by performing walkdowns of existing equipment where possible. In some cases, operators performed step-by-step procedure walk-throughs to validate procedural methods. Operators and System Managers worked together on each procedure. Identification of expected plant responses were verified. Proper identification of Safe Shutdown Methods and alternate cooling safety functions were verified. Second ten-year interval IST requirements were incorporated per current ASME Code.

Each procedure was assured to be technically correct using current references. Additionally, the process focused heavily on walkdowns, interfaces with other procedures, and the practical experiences of Operators. When inconsistencies were identified between technical documentation and plant hardware, the problems were resolved using the appropriate corrective action process.

The procedure rewrite project facilitated an integrated assessment of the plant's systems, structures, and components ability to function as described in the procedures. The System Manager and Operator walkdowns provide reasonable assurance that the actual plant configuration enabled the functions described in the procedures and that plant response is consistent with the procedures.

2. Peach Bottom

a) Peach Bottom System Operating (SO) Procedure Rewrite

In a one-time project between 1987 and 1989, the existing PBAPS operating procedures were replaced by regenerated unit specific procedures. Each procedure was drafted with the use of available technical resources and was verified against Technical Specifications and controlled prints. Additionally, the draft process included walkdowns, information from the existing procedures, and interviews with operations and system engineering personnel. The project included a documented verification and validation process that included a simulated, in-plant execution of the procedure. The SO rewrite project involved a systematic regeneration of all the operating procedures, and therefore, the validation process involved a systematic verification of the hardware of much of the operationally functional components in the plant. The rewrite project enabled an integrated assessment of the plant's systems, structures, and components ability to function as described in the procedures. The operator walkdowns involved a verification that the actual plant configuration was consistent with the functions described in the procedures.

b) Peach Bottom Surveillance Test (ST) Procedure Rewrite

The ST rewrite project at PBAPS involved a one-time replacement of the existing surveillance tests, tests associated with inservice inspection, the Fire Protection Plan, and routine tests. The project took place between 1990 and 1992 and was part of the response to an NRC commitment. The rewrite included a screening process to identify surveillance requirements contained in the Tech Specs and other regulatory commitments, and test frequencies were verified. The technical content of the tests considered the most recent design considerations and controlled plant prints. A verification and validation program included reviews by system engineers and end users. Walkdowns were performed for usability. The procedure approval process requires 10CFR50.59 Reviews. The comprehensiveness of the rewrite including verification and validation demonstrated an understanding of the actual plant equipment versus the design. The plant walkdowns did not verify every design characteristic of the affected plant equipment, but the systems, structures and components were examined for lineup and function. Verifying that the testing matched the Tech Spec surveillance requirements provided a bench mark assurance that the performance and capabilities of equipment important to safety would be adequately monitored.

J. Peach Bottom Critical Equipment Monitoring System (CEMS) and Plant Labeling

The CEMS project, which started in 1981 as a PECO Nuclear initiative, provided a consistent labeling method for piping system valves and 480 V motor control units at Peach Bottom and installed labeling on those components. Completion of the CEMS and an additional project to label important plant components such as pipes, motors, fans, etc. not in the scope of the CEMS project, later became a commitment as a result of INPO and NRC inspections that identified labeling weaknesses. Both projects were completed by June of 1990. The projects were accomplished using controlled plant prints to initially identify the equipment, and system walkdowns were performed to further clarify and in some cases correct the as-built understanding of the existing components. The established identifications were added to the P&ID's and other controlled documents such as station procedures. The CEMS and labeling projects were valuable configuration management activities because of the comprehensive system walkdowns which were performed. Additionally, the project established a baseline tool, in the form of unique component identifiers, to be used by all configuration management processes. The understanding of the as-built configuration provided by the projects continues to enable performance assessments.

K. Limerick Electrical Separation Walkdowns

The purpose of the Limerick Electrical Separation Verification initiative was to walkdown, identify, verify, and resolve identified concerns within Limerick electrical panels. This initiative consisted of a review of Licensee Event Reports (LER) generated from 1988 which were related to electrical separation concerns, internal PECO Nuclear assessments and associated walkdowns along with external contractor assessments/walkdowns.

A review was completed during November and December of 1990. This review encompassed a walkdown and assessment of 100% of the Unit 1 panels. Plant management made the determination that 100% of the deficiencies were to be reworked and completed prior to restart of Unit 1, which was shutdown for the third refueling outage. A review and walkdown of Unit 2 panels was conducted, completed and reworked during the Unit 2 first refueling outage.

Once the baseline inspection/verification of the panels was completed, an ongoing assessment and random sample verification was performed by the QA Technical Monitoring Branch on a monthly basis through the first quarter of 1995. Currently, a surveillance is performed by station NQA personnel of a random sample of panels, once per year, and as applicable during surveillance of Maintenance/I&C activities, if the activity involves a panel which has separation criteria. The NQA surveillances have not identified additional concerns.

Formal Root Cause Analyses were completed in 1989 and 1991. Some of the lessons learned from these analyses, which were subsequently implemented, included additional training for installation, inspection and engineering personnel around the requirements and acceptance criteria associated with electrical separation criteria. This was completed utilizing formal training programs for impacted station personnel, procedural reviews and posted information. Additionally, procedures required revision to clearly integrate separation criteria. These procedures included those used in maintenance, operations, engineering and installation activities. A field inspection checklist was developed to assure common, consistent acceptance criteria. Finally, the station modification and engineering change process was revised to include a review of possible impact (positive or negative) of the electrical separation criteria. The latest Electrical Work Products Assessment (an internal assessment completed in 1996) identified the need to enhance the current instructions at Peach Bottom to include specific criteria in the area of electrical separation. From this assessment, PBAPS specification NE-00256 has been created to specify the separation requirements.

All corrective action activities associated with the lessons learned were incorporated in the applicable training material and the applicable procedures. All identified discrepancies with hardware were resolved and corrected to comply with applicable design requirements. If additional concerns are identified, resolution would be completed utilizing the various corrective action processes, i.e., the Engineering Change Request (ECR) process utilizing common procedure MOD-C-9 and the PEP process using procedure LR-C-10. The review, inspection and resolution processes associated with the electrical separation initiative created a baseline of data. Corrective actions have strengthened barriers in the modification and work process which requires an assessment of electrical separation impacts, during current and future design changes.

L. PBAPS Thermal Overload Verification

The scope of this program involved an assessment and walkdown verification of all Motor Control Center (MCC) thermal overload relay devices on safety related electrical circuits. The program was implemented based upon an internal PECO Nuclear evaluation, SSFI findings and concerns, spurious tripping/operation of plant equipment and numerous nonconformance reports requesting resizing of the existing thermal overload devices. The program involved walkdowns of all MCC compartments, walkdown of the safety related motors for verification of motor nameplate information and data, development of a thermal overload sizing guideline, resizing the relays, field work, as required, and revisions to the appropriate design documentation and drawings. The bulk of the program was completed from June 1993 through 1994. The guideline developed has been incorporated into a common procedure, NE-C-301, and is utilized by design engineering for on-going verification of proper thermal overload protection of the motor circuits.

Based on in-plant walkdowns, subsequent NCRs generated, and corrective actions taken where problems were found, there is reasonable assurance that the plant configuration is consistent with the design bases in this area.

M. Service Water System-Related Recalculations

The design bases heat loads for the Emergency Service Water (ESW), Residual Heat Removal Service Water (RHRSW) (LGS only) and High Pressure Service Water (HPSW) (PBAPS only) systems at both Peach Bottom and Limerick have been regenerated in the last five years using as-built plant configurations. The ESW system heat loads have been recalculated through the use of a computer code which allows transient modeling of the reactor buildings. This code also allows transient modeling of the response of the Emergency Core Cooling System (ECCS) pump room unit coolers. This model is controlled by two calculations at each site and is updated to reflect changes affecting the heat loads in any of the ECCS rooms. The RHRSW and HPSW heat loads have been recalculated using ANS standard 5.1 to more accurately reflect plant decay heat loads. All plant heat loads have been recalculated to account for the effects of Power Rerate which was previously discussed.

Additionally, computer flow models exist for both the Peach Bottom and Limerick ESW systems and the Limerick RHRSW system. The Peach Bottom HPSW system is a simple once through system and, therefore, is not computer modeled. These models predict system flows under quasi-steady state conditions. These models are used to predict the effects of piping degradation and system modifications, and are used in system flow balancing. These models have also been updated to reflect modifications to the ESW systems. Using these two codes, PECO Nuclear performs accurate assessments of heat exchanger performance under varying accident conditions with varying heat sink temperatures and system flows. These codes are also used in the assessment of routine heat exchanger test data.

The ultimate heat sink at Limerick has also been computer modeled within the last two years to provide a more accurate assessment of spray pond performance with regard to heat removal and inventory loss under design bases accident conditions. This model is also a controlled design calculation. This model was used to assess the effects of Power Rerate and is routinely used to make operability determinations.

The design calculation for the performance of the cooling towers in the Emergency Cooling Water (ECW) System at Peach Bottom was reperformed between 1990 and 1992. A special test procedure was also performed to demonstrate that the system can perform its design function.

N. High Energy Line Break (HELB) Analysis Program

In 1991, PECO Nuclear implemented a program to control all HELB and Secondary Containment barriers. Prior to this time, only Fire Barriers were in the penetration control program. The program included a walkdown of all existing HELB barriers and Secondary Containment barriers to verify their status. These walkdowns continued during 1991 and 1992. NCRs were written to resolve discrepancies by either revising the analysis or upgrading the barriers. Based on the walkdowns and subsequent calculation and field rework, there is reasonable assurance that this portion of the design bases is consistent with plant configuration. This program is ongoing in that the analysis is continually updated as affected by modifications or other barrier work.

ATTACHMENT 1

O. Setpoint Methodology Program

From December 1989 through December 1994, PECO Nuclear developed a setpoint methodology control program at PBAPS. A similar program was already established for LGS prior to 1990. The Setpoint Control Program regenerated Technical Specification, Important to Safety, and Reliability Centered Maintenance (RCM) instrument setpoints at PBAPS, involving approximately 200 calculations. In addition to regeneration, performance information is regularly obtained from periodic surveillance testing. The setpoint control software program is in place and used on a continuing basis which assures compliance remains strong. This program provides reasonable assurance that the plant design bases is consistent with the plant configuration for the instruments involved.

P. Peach Bottom/Limerick TRIP Procedure EPG Rev 4 Upgrade

This program involved the incorporation of the BWR Owner's Group Emergency Procedure Guidelines (EPG), Rev. 4 into the Peach Bottom and Limerick emergency operating procedures (EOP). The upgrade resulted in a complete rewrite of all the Transient Response Implementation Plan (TRIP) procedures which became effective in the 1990-1991 time frame. The generation of the procedures was performed using a rigorous process that documented a step for step justification of the plant specific version of the generic EPG step. This involved obtaining design information from controlled plant documents and performing calculations using approved methods. All procedures were validated by technical reviews and in-plant walkdowns or simulation. The 10CFR50.59 review was made against the full extent of the SAR including the UFSAR and the SER associated with revision 4 of the EPG's. Because the EPG's provide the best overall symptom based guidance for mechanistically possible events there exists some differences in the design approach to equipment operation and the EPG approach. A review of the differences is now documented in the UFSAR itself. The comprehensiveness of the reviews associated with writing the procedures and walking them down gave an opportunity to verify the relationship between the lineup and function of safety related systems, structures, and components. The rewrite of the EOP's benchmarked the expected response of equipment and operators to the license and design commitments.

Q. Design Baseline Document (DBD) - System Testing Validation

As part of DBD preparation activities, a review was performed to verify that all system design bases controlling parameters identified in the DBDs had been or were being tested in the plants. Approximately 20 to 30 station procedures per system were reviewed to verify that tests of the controlling parameters utilized acceptance criteria consistent with the DBD values. Action items were generated and dispositioned for discrepancies between the Station Test Procedure acceptance criteria and the DBD values or where no test of a DBD controlling parameter could be identified. These activities support the rationale that system design bases performance requirements (parameters identified in the DBDs) and the physical abilities of plant systems to fulfill these requirements are consistent and that the plant is capable of performing to its design bases.

II. ADDITIONAL VERIFICATION PROGRAMS/PROJECTS/INITIATIVES

In addition to the above programs, there have been numerous initiatives at Limerick and Peach Bottom that have provided valuable opportunities to compare plant hardware and performance to the design bases. These programs have been varied in purpose and scope, but when they are considered together, they provide reasonable assurance that SSC configuration and performance matches the design bases today. Unless otherwise noted, the programs are common to both Limerick and Peach Bottom. These programs have in some cases provided actual walkdowns of plant components for verification purposes or in other cases correlated the performance of SSCs to design requirements. The following examples are noted.

ATTACHMENT 1

- PBAPS Restart Program

  A comprehensive assessment of configuration management issues was conducted in the following functional areas: procurement, modifications, drawing control, maintenance, operations, licensing, training, QA/QC controls, testing, security, and design bases.

- PBAPS Maintenance (M) and Instrument & Controls (IC) Procedure Upgrade

  Maintenance and I&C procedures were rewritten in a one-time project between 1989 and 1991. The procedures incorporated information contained in prints, vendor manuals, commitments, LGS procedures, regulations and equipment history.

- Individual Plant Examination of External Events (IPEEE) Project

  IPEEE walkdowns to identify severe accident vulnerabilities if any, and report results for five external events (seismic, internal fire, tornado/high wind, external floods, and transportation accidents).

- Environmental Qualification Program

  Components included within the Environmental Qualification program for LGS & PBAPS were field verified for make, model, configuration details (LGS only), and location. Approximately 90% of all Environmental Qualified components outside of primary containment and 50% of all Environmental Qualification components inside primary containment have been field verified.

- Modification Process Design Walkdowns

  Plant walkdown and reconciliation between as-designed and as-built conditions is required for plant modifications.

- Electrical Panel Loading / Station Blackout Calculations

  A verification of the as-built electrical panel loads at both stations was performed. From this data, the electrical loading calculations were generated/revised.

- Technical Specification 24 month Fuel Cycle Change

  A review was completed to verify that the as-built plant associated with the 24 month fuel cycle change met the required design criteria based upon an extended instrumentation calibration frequency.

- PBAPS Control Room Human Factors Enhancement

  The control room design review and enhancement benchmarked the control room configuration aspect of design. Modifications verified performance of controls and indications by acceptance testing. The design review and recommendations optimized the ability of the systems to be used by operators in a manner assumed by design and license commitments.

- Flow Accelerated Corrosion (FAC) Program

  FAC Program identified and evaluated piping routes which are susceptible to FAC. A review of piping design documentation and any posted change paperwork was performed. Walkdown of the applicable systems and piping was completed, as required. Overall, the program identified no adverse conditions with regard to the design of the piping systems included in the program.

• PBAPS Seismic Qualification Utility Group (SQUG) Findings	Safe Shutdown Equipment List prepared in response to GL 87-02 required extensive document review and walkdown to identify valid safe shutdown paths.
• Nuclear Fuel Configuration	PECO Nuclear's QA assessments focus on GE's fuel design and fabrication activities unique to the upcoming core reload. In this way, confidence is gained that the product provided and analysis performed is pertinent to the designed reload/plant conditions.

III. CONCLUSIONS

PECO Nuclear has determined that there is reasonable assurance that the configuration and performance of the Peach Bottom and Limerick station's systems, structures and components (SSCs) are consistent with the design bases. This determination is based on an extensive assessment of past and existing PECO Nuclear programs and processes where design bases review was the primary purpose, plus other programs or tasks where design bases verification was a benefit of the activity. The programs reviewed generally included physical walkdowns to verify the SSC versus design bases information.

There have been several comprehensive programs which have compared existing plant configuration with its associated design bases documentation. The multiple PECO and NRC SSFIs, and the recent UFSAR Verification Program at both stations were broad scope activities which verified configuration consistency. The power rerate project and heat load/flow modeling of the cooling water systems regeneration project both verified and regenerated the affected systems design bases via plant modifications and calculation revisions. The other programs and processes identified in this summary have independently verified the configuration and performance of specific systems, components or attributes. Cumulatively, these activities provide reasonable assurance that the overall SSC configuration and performance are consistent with the plant design bases.

A review of the aggregate SSFI results has identified a potential concern relative to past calculation control. A PEP issue has been generated to investigate this potential concern and to implement necessary corrective actions.

The UFSAR Verification effort identified process enhancements. Recommendations included implementing a guideline for administrative control and maintenance of the Personal Librarian Software (PLS) database and overall changes to the ownership of the UFSAR sections. A PEP issue has been generated to evaluate and identify any Conditions Adverse to Quality (CAQ), root causes, generic implications and corrective actions through a root cause analysis. The PEP process will also track the previously listed recommendations to closure. PECO Nuclear is committed to complete the verification of the UFSAR and is evaluating the related scope and schedule. Additional information will be provided in future correspondence (see commitment 1 in Attachment 3). Closure of identified discrepancies continues. Corrective actions associated with the program concerns will be identified and tracked via the PEP process once the review is completed. Additional baseline and continuing training of nuclear personnel will provide continued focus on UFSAR content, accuracy and fidelity.

ATTACHMENT 1

REQUEST (d)

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

RESPONSE

Issues are identified and corrected by the variety of problem identification and/or corrective action processes that exist within PECO Nuclear, including the process for the Control of Nonconformances (NCRs). Conditions Adverse to Quality (CAQ) are processed by the Performance Enhancement Program (PEP). Procedure LR-C-10, Performance Enhancement Program (PEP), defines a Condition Adverse to Quality as "A condition where procedures, work processes, or activities permit the potential for, contribute to, or result in failures, malfunctions, deficiencies, deviations, defective material or equipment, or noncompliance with specified requirements." The PECO Nuclear PEP process is utilized for the identification of problems and implementation of corrective actions, including actions to determine the extent of problems and actions to prevent recurrence, fulfilling the requirements of 10CFR50 Appendix B Criterion XVI. The PEP process and other problem identification and/or corrective action processes are discussed below. In addition to the PEP and NCR processes, it is important to understand the key attributes of an effective problem reporting and resolution program. Several of these attributes including the escalation process for problem resolution, the operability and reportability processes, the routine interface with the NRC and the processes for continuous learning opportunities through generic NRC guidance and industry events are also discussed. Generic and specific training is also identified. The discussion of these processes also includes an evaluation of performance based on external and internal reviews. The combination of effective problem identification and corrective action processes, an appropriate operability and reportability decision processes, and a structured external event evaluation process supports the continuous improvement culture at PECO Nuclear and provides reasonable assurance that problems are identified and resolved.

I. PROBLEM IDENTIFICATION AND CORRECTIVE ACTION PROCESSES

A. Performance Enhancement Program (PEP)

LR-C-10 is the PECO Nuclear procedure that establishes the Performance Enhancement Program (PEP) to improve performance through evaluation of conditions adverse to quality and other enhancement opportunities, including trend information. The procedure provides direction for the identification and evaluation of issues to ensure they are thoroughly reviewed, including causal factor and generic implication identification with subsequent implementation of corrective actions to prevent recurrence. LR-C-10 also establishes a uniform approach for identification and evaluation of potentially reportable items.

A PEP issue is initiated when a condition adverse to quality, a problem, a potential problem, a Potentially Reportable Item (PRI), a concern, an undesired occurrence, or a near miss occurs for which the resolution provides an opportunity to enhance performance. Guidance is provided in LR-C-10 as to what types of issues should be entered into PEP. PECO Nuclear employees and contractors who become aware of issues during the performance of normal job duties, are responsible to enter such issues into PEP. The PEP process recognizes that there are many problem identification and/or corrective action processes within PECO Nuclear that are sufficient in and of themselves.

A PEP issue is generally not warranted for issues contained within other problem identification and/or corrective action processes; however, LR-C-10 is applicable and in effect at all times. An issue from one of these other processes may escalate into a PEP issue. This escalation occurs as the result of an issue having significant consequence or a pattern of recurrence that warrants more formal evaluation.

Upon identification of a PEP issue, an assessment of the need for immediate corrective actions and communication of the issue to a Reportability Coordinator (RC) for assessment of applicable reporting requirements is performed. Immediate corrective actions are implemented as necessary and the RC, Shift Management, and the Regulatory Engineer perform the necessary notifications.

PEP issues are then reviewed and classified by an Experience Assessment Coordinator (EAC). Depending on the significance and consequence of the issue and whether a similar PEP issue exists, the EAC assigns an appropriate Significance Level of 1, 2, 3, or 4. The EAC also assigns an Evaluation Class of A, B, or C depending on the extent of issue investigation/analysis deemed appropriate. The most significant issues are level 1; the most rigorous evaluations are class A. The issue investigation is then the responsibility of an assigned cognizant organization. For significant PEP issues, use of a multi-discipline team is considered.

The ensuing investigations are based on objective evidence. Root cause analysis techniques, which may include event and causal factor charting, barrier analysis, change analysis, and task analysis, and the subsequent assessment of generic implications are used to identify the extent of the problem. Corrective actions to prevent recurrence are defined and implemented to address the identified causal factors and generic implications. Interim corrective actions are established, as necessary, to assure adequate performance until completion of final corrective actions.

Upon completion, corrective actions are reviewed to assure that the intent of the corrective actions has been met. For the more significant or consequential events or as directed by management, documentation of a corrective action effectiveness review is required. Tools used for corrective action effectiveness determination include sampling, observation, performance indicators, and determining what constitutes acceptable performance and assuring that it has been met.

In addition, the Experience Assessment Coordinator (EAC) performs analysis of the PEP data to identify any adverse or positive trends. This review takes into consideration root causes, significance, keyword, and identification methods. Recurring issues are also identified. This information is provided to PECO Nuclear management and the Nuclear Review Board (NRB). Furthermore, on a monthly basis, a summary of the number, significance, level, and age of open issues is presented to management. The status and significance of any overdue evaluations and corrective actions are presented.

B. Control of Nonconformances

A-C-901, Control of Nonconformances, is the PECO Nuclear procedure that establishes a uniform approach to identify, document, evaluate, and resolve hardware nonconformances found in installed plant equipment using Nonconformance Reports (NCRs). A-C-901 addresses deviations from the licensing or the design bases as described in the Safety Analysis Report (SAR) or situations requiring field changes.

All PECO Nuclear personnel and contractors are responsible to initiate an NCR upon identification of apparent nonconformances. The initiator provides a detailed description of the identified condition. The initiator and the initiator's supervisor are responsible for initial operability and reportability determinations. Operations, Engineering and Reportability Coordinators (RCs) are consulted as necessary in completing initial operability and reportability determinations. Shift management is immediately notified of potential operability concerns. A RC is immediately notified of potential reportability concerns (Further discussion on operability and reportability determinations is provided in Section II, below.)

The NCR is assigned to a cognizant organization/individual for dispositioning. The evaluator determines if the nonconformance is dispositioned as Rework, Repair, or Use-As-Is. All field activities necessary to resolve Rework and Repair dispositions (i.e., corrective actions) are identified. A final operability and reportability determination is made, including providing engineering justification for the determination. Operability impact is based on the ability of a system, structure or component to perform its specified design function. The evaluator consults with the System Manager and Station Operations Management as appropriate.

Beginning in September 1995, NCRs are processed under the Engineering Change Request (ECR) (MOD-C-9) process. This revision provided simplification of processes and procedures. As such, requirements for configuration management are inherently incorporated into NCR dispositions and subsequent field activities and document revisions, including procedure impact reviews. This includes expectations for involving personnel from other organizations to assure proper incorporation of their discipline/expertise. 10CFR50.59 Reviews are performed for Repair and Use-As-Is dispositions.

If a final disposition cannot be determined, requirements are identified for providing an interim disposition, including documentation of the time period for which the interim disposition is acceptable.

An NCR can result in the initiation of a PEP issue for a Potentially Reportable Issue (PRI), a revision of the operability determination to inoperable during the performance of the disposition, a revision of the reportability determination to reportable during the performance of the disposition, or an NCR for which there is a Condition Adverse to Quality (CAQ).

C. Quality Concerns and Allegations

Procedure A-C-905, Quality Concerns & Allegations, addresses how an individual's quality concerns can be reported, and identifies alternatives available to an individual who believes that their previously reported quality concerns are not receiving proper consideration. A-C-905 describes the process for the control, resolution and documentation of quality concerns and allegations. This procedure applies to quality concerns identified by PECO Nuclear employees and contract personnel associated with activities for PBAPS and LGS, and to allegations received through the NRC or other outside sources. Personnel are made aware of this procedure and process via General Employee Training and by Nuclear Quality Concern posters which are conspicuously posted at various locations in the nuclear headquarters and at the two nuclear stations. The procedure and the Nuclear Quality Concern posters provide the contact phone numbers for the Quality Concerns Hotline and the NRC. The procedure also states that individuals who are not satisfied after following the above process, or at any time, may contact the NRC. On a quarterly basis, the NQA Director provides a status of the Quality Concerns to the PECO Energy Chairman of the Board and the Nuclear Committee of the Board of Directors.

D. Training

General Employee Training (GET) is required for personnel with unescorted access to PECO Nuclear plants. In the Quality Program portion of GET, there is information on the responsibilities of individuals when they discover a deficiency during their normal work processes. Responsibilities identified include:

- notification of supervision
- use of Equipment Trouble Tags (ETT) AG-CG-26.1

- use of the PEP process
- awareness and use, as appropriate, of the Quality Concerns Hotline.

These provide assurance that deficiencies are identified and properly processed per PECO Nucle established procedures for corrective action.

Additional training is provided to designated engineering personnel as part of the Engineering Support Personnel (ESP) orientation manual and System Manager Qualification Manual. The orientation training focuses on providing awareness of the governing procedures and defining the relationship of these procedures to job function. The qualification manual requires demonstration of skills for NCRs and PEP issues, including root cause analysis, dispositioning of issues and processing of these items in Plant Information Management System (PIMS) prior to performing these tasks independently.

Root-cause analysis is a fundamental characteristic of an effective corrective action program. Formal root cause analysis training is provided to selected PECO Nuclear personnel. Significant PEP issues require that a formally trained individual perform the root cause analysis.

E. Summary of Performance

The adequacy of PEP is self-assessed annually. These annual self-assessments are supplemented by the periodic performance indicators and trend reports that are developed and distributed to management. On an ongoing basis, the health of the program is also considered during the periodic Experience Assessment Council meetings and as a result of feedback from station management, senior management and the Nuclear Review Board. As a result of self assessment activities, in 1995 a change to the process was made to formalize the mechanism for the corrective action effectiveness determination prior to closeout of a PEP Issue. Significance Level 1 and 2 PEP issues are required to have a corrective action effectiveness determination. Exemption from formal corrective action effectiveness determinations for Significance Level 3 PEP issues must be justified, documented and approved by management. This change was incorporated into a revision and was effective in December 1995. The line organizations continue to gain experience in performing corrective action effectiveness evaluations. It is recognized that these evaluations, since they are a new requirement, require oversight which is occurring by NQA surveillances and assessments, by the Experience Assessment Council activities, and by Engineering Assurance assessments.

Internal oversight of the PEP program is performed by NQA assessments and surveillances and ISEG reports. Overall, the implementation of the PEP process has been found to be a strength. Some problems and potential enhancements have been identified and appropriate actions have been taken. Specific areas where this was noted were the areas of root cause analysis, generic implications and effective corrective actions to prevent recurrence.

In 1995, NRC inspections were conducted at Limerick and Peach Bottom regarding problem identification and resolution. In each of the reports, the PEP process was positively noted. The threshold for the identification and documentation of issues was determined to be appropriate, as was management oversight and involvement. At Limerick, the inspection noted several examples of repetitive problems where interim corrective actions were ineffective and final corrective actions were not timely or effective. It was noted that PECO Nuclear had identified most of these weaknesses, taken strong measures, and applied resources to improve performance in the affected area and corrective action process. Overall, the inspectors concluded that the quality of PEP issues reviewed was good.

Another independent evaluation of the Corrective Action Program and specifically the PEP process was completed in November 1996 as part of the 1996 Joint Utility Management Audit (JUMA) of

PECO Nuclear NQA Department. The PEP process was concluded to be well defined and properly implemented. Overall effectiveness was determined to be adequate.

NRC inspection reports (95-80/80) for LGS and PBAPS included discussion of the processes for control of nonconformances. The inspectors concluded that PECO Nuclear's nonconformance report process was effective in identification and dispositioning of deficiencies. The revision to govern NCRs by the ECR process, implemented in September of 1995, was a self identified improvement to simplify the process and procedures. The most recent NQA surveillance identified some deficiencies for which corrective actions have been implemented. The deficiencies were identified as not significant in terms of impact on the design bases and operation of the two stations. Recent self assessments (1996) identified strengths and weaknesses with the NCR process. The training bulletin issued for the NQA identified deficiencies also addressed the self assessment results. NQA and line organization follow-up to date have identified the corrective actions as effective.

NRC inspection reports (95-80/80) for LGS and PBAPS also included discussion of the employee concerns program as a part of the problem identification and resolution inspections conducted at each location in 1995. At LGS, the NRC inspectors concluded that the Quality Concerns Hotline program was not well known by plant personnel, and the inspectors noted only one posting during their tours of the facility. At PBAPS, the NRC inspectors found that the level of knowledge of the program varied. The inspectors concluded that employees were aware of avenues available to them for raising quality concerns and that a very good open atmosphere for communication up and down through the organization exists at PBAPS. In each of the two NRC inspection reports, it was noted that the number of calls to the Quality Concerns Hotline had decreased and that the number of calls received on the Hotline indicated that the program was utilized. An indication of the awareness and the effectiveness of the Quality Concerns Hotline and PECO Nuclear management practices can be inferred from NRC data pertaining to allegations. Based upon a review of the NRC memo dated 10/7/96, subject: Annual Report of the Allegation Advisor, PECO Nuclear determined that the number of allegations for the last 2 years for the PECO Nuclear facilities was relatively low compared to the other facilities listed and the PECO Nuclear facility trend was towards lower numbers. This trend was believed to be notable since the NRC report stated that the total number of nuclear utility allegations for fiscal year 1996 (for the first 8 months) was significantly higher than fiscal year 1995. To address the employee awareness issue, actions taken by NQA included increasing the number of Hotline postings, confirming that the Hotline was a topic in General Employee Training, and communicating the existence of the Hotline to management and station personnel.

II. OPERABILITY AND REPORTABILITY DETERMINATIONS

A. Operability Determinations

The corrective action processes described above include the need to make operability determinations on Systems, Structures, and Components (SSCs). Operability is initially evaluated by personnel initiating and dispositioning deficiencies per guidance provided in process specific procedures (i.e., A-C-901 and AG-CG-3). Per A-C-901 consultation with operations/shift management is required for potential operability concerns. LR-C-10 requirements for processing of a PEP issue involve Shift Management. Ultimately, operability determinations are the responsibility of Operations/Shift Management. In addition, the Reportability Coordinator (RC) reviews the event for potential equipment operability concerns. If any equipment is potentially inoperable, the RC promptly notifies Shift Management. Shift Management review is required for all issues that are potentially reportable or involve potential equipment inoperability. Shift Management makes the final determination of equipment operability and takes appropriate compensatory action for inoperable equipment.

In addition to the Technical Specifications, the Operations Manual provides guidance to operators for determining operability.

Additionally, the Operations Services Section is responsible to review daily planning for work and testing activities, developing plans for timely execution of work windows addressing unmet Limiting Condition for Operations (LCO) and assessing work schedule impact for plant safety.

B. Reportability Determinations

The corrective action processes described also require determination of NRC Reportability (e.g., 10CFR50.72, 10CFR50.73, and 10CFR21.21) of the issue. LR-C-10 and the referenced Reportability Reference Manual (RRM) provide guidance to PECO Nuclear employees in identifying potentially reportable items, and performing preliminary and final reportability determinations. The RRM contains a tabular index, sorted by subject, of event driven, non-routine reporting requirements applicable to the operation of PBAPS or LGS. In addition, a discussion of each requirement is provided to assist in interpretations of requirements. The RRM is intended for use by the Reportability Coordinators (RCs), Regulatory Engineers (REs), Shift Management and others as guidance in performing reportability related activities as defined in LR-C-10.

The RC reviews the event and determines if the item is a Potentially Reportable Item (PRI) using the Reportability Reference Manual (RRM). Shift Management makes a determination whether a prompt notification is required using the RRM and assures all prompt notifications are made in the manner and time limit specified by the applicable reporting requirements. Prompt notifications are made using the Emergency Notification System (ENS) network.

The Regulatory Engineer (RE) reviews PEP issues for reportability and notifies the responsible management and individuals when a verbal notification (non-prompt) is necessary. The RE assures that notifications are retracted or revised, if necessary. The RE ensures the preparation of the draft non-routine report is in accordance with regulatory requirements and obtains PORC review of the non-routine reports.

Appropriate reporting is a critical attribute of a successful organization. There are several barriers to ensure that appropriate reporting is maintained. The Reportability Coordinator (RC) reviews the event to determine if the item is a Potentially Reportable Item (PRI). If any equipment is potentially inoperable, the RC promptly notifies Shift Management. Shift Management review is required for all issues that are potentially reportable or involve potential equipment inoperability.

Finally, the RE reviews all issues for reportability and notifies the responsible management and individuals when a verbal notification (non-prompt) is necessary. As shown by the information presented above, the reportability process at PECO Nuclear, including the initial reportability considerations by the RC in conjunction with the final reportability determinations by the RE, serves to assure adequate and timely reporting to the NRC.

C. Training

Both PBAPS and LGS have documented training courses for the Reportability Coordinators (RCs). A review of RC training qualification records indicates that individuals in a broad cross-section of Departments at the stations and the PECO Nuclear Headquarters have successfully completed RC training.

A Qualification Manual is in place for the Regulatory Engineer (RE) which provides directions for training and evaluation of their PEP review function. Once Reportability Reference Manual Training is completed, the trainee RE is evaluated for satisfactory performance of the task prior to performing the task independently.

D. Summary of Performance

1. Operability

Based upon review of recent NQA assessment reports, operability determinations have been properly made at LGS and PBAPS. Open limiting condition for operations (LCOs) and potential limiting condition for operations (PLCOs) were reviewed and determined to be properly identified and controlled in accordance with applicable Technical Specifications and the Operations Manual. Closed LCOs and PLCOs were reviewed and NQA determined that the closure mechanism was correct and adequate for establishing operability of the system or component. Based upon a review of NRC inspection reports for the last two years and a review of NRC violations since 1989, operability determinations have been generally noted to be properly made. For specific cases, where inoperable systems or components were identified or operability determinations were determined to be in error, actions were taken which resolved the situation. As a result of internal operating experience and NRC Generic Letter 91-18, PECO Nuclear has improved Licensed Operator training and Operations Manual guidance regarding operability determinations.

2. Reportability

A review of NRC, Industry and Internal PECO Nuclear assessments at PBAPS and LGS since 1990 was performed. NRC inspection reports indicated that reportability determinations, in general, were properly made although two reportability issues, one at each station, were identified. The LGS issue occurred in June of 1993 when two late four-hour notifications were made to the NRC pursuant to 10 CFR 50.72. Plant personnel conducted an investigation of the two events, and identified the need for some corrective actions. The inspectors reviewed the late notifications and the corrective actions, and concluded that the corrective actions taken and planned were adequate to address the identified deficiencies. The PBAPS issue occurred in June of 1995 when an NRC inspector noted possible areas for improvement regarding the timeliness of the reportability determination associated with two emergency diesel generators being inoperable at the same time.

With respect to NQA assessments, various aspects of reportability are periodically reviewed during Corrective Action audits, as appropriate. A review of these audits indicated that reportability determinations were appropriately made. With respect to ISEG assessments, these assessments typically do not focus on reportability. One exception was an ISEG assessment (ISEG 93-121) conducted at LGS in late 1993. This assessment focused specifically on NCR operability and reportability determinations, and identified only two minor reportability concerns that were subsequently addressed. Overall, review of the issues identified above revealed that no significant changes were made to the reportability process in response to these issues. Reportability is reviewed during the periodic station Self-Assessments.

III. NRC INTERFACE

In addition to the reportability process, other interactions and communication between PECO Nuclear and the NRC occur in various ways, for example, the communication with the resident inspectors at PBAPS and LGS. Such communication is a natural outgrowth of the ongoing inspection activities conducted by the residents, and takes on several forms, e.g., NRC attendance at station sponsored meetings, or discussions with station personnel during observation of station activities. The Daily

Leadership Meeting is typically attended by the NRC resident inspectors and provides an opportunity for the Station Management to assure that the NRC resident inspectors are well informed of significant issues and ongoing station initiatives. Any issues or concerns raised during this meeting provide follow-up opportunities for the residents directly with the individuals or organizations that are involved.

IV. EVALUATION OF EXTERNAL EVENTS

A. Operating Experience Assessment Program

Linked with the corrective action and reportability processes is the Operating Experience Assessment Program (OEAP), which is used to evaluate external information. This program evaluates information from the nuclear industry and from the sharing of in-house events between the nuclear stations to determine the actions needed at PECO Nuclear to mitigate or prevent similar events from occurring at PBAPS or LGS. The process also provides information to various groups to keep them informed of operating experience. The process requires documented evaluations and formal responses for the more significant information. If it is determined that changes are required to PECO Nuclear programs/procedures, then changes are made in accordance with approved procedures. If appropriate, these changes may also be annotated.

B. Training

A Qualification Manual is in place for the OEAP Coordinator which provides directions for training and evaluation of this particular task. This individual must be satisfactorily evaluated on performance of a task prior to performing the task independently. The trainee must screen OEAP, Bulletins, Notices, and other incoming documents to determine departmental responsibilities and maintain status of each item in accordance with LR-C-01 "Commitment Tracking Program" and LR-C-04 "Operating Experience Assessment Program". OEAP process training is provided by various means. For example designated new engineers are trained in the OEAP process via an approved training module. This is required per the Engineering Support Program (ESP) Training. In addition, continuing training contains discussions associated with major operating experience issues both in-house and throughout the industry.

C. Summary of Performance

A review of NRC, Industry and Internal PECO Nuclear assessments at PBAPS and LGS since 1990 was performed. With respect to NRC inspections, two NRC inspections specifically included reviews of the OEAP process. These inspections indicated that operating experience was effectively evaluated and implemented at both sites. As an example, LGS review of industry experience prior to the September 11, 1995, inadvertent Main Steam Relief Valve (MSRV) opening event focused on the specific issues identified in various industry and regulatory documents regarding ECCS suction strainer clogging, and determined such issues were either already addressed or they were not applicable to LGS. The belief at the time, consistent with the focus of the industry and the NRC, was that the fibrous material of concern was drywell insulation destroyed in a Loss-of-Coolant Accident (LOCA) and transported to the suppression pool, and not a pre-existing condition in the suppression pool. During the special inspection following the September 1995 event, the NRC reviewed LGS responses to NRC Bulletin 93-02 and its supplement, and determined, as indicated in NRC Special Inspection Report 95-81/81, that the responses were adequate.

Regarding the Industry Plant Evaluations, ISEG, and NQA assessments, the results were mostly positive with some minor exceptions. An NQA Performance Assessment Division (PAD) assessment performed in 1990 did identify several areas for enhancement (e.g., procedures and process lack commonality, greater efforts should be made to share ideas). The recommendations of the PAD assessment have either been completed or superseded by new processes. An Operating Experience task team was organized to review the utilization of operating experience at the stations. As a result of this review, several recommendations were made to enhance the OEAP. The OEAP process and its effectiveness are reviewed during the periodic station Self-Assessments.

Many significant improvements have occurred over the past few years in the area of operating experience. These improvements include the utilization of the INPO Nuclear Network information at the morning leadership meetings at both stations. This action allows the station management to react promptly to industry issues. Another enhancement for PBAPS occurred during the 1987 shutdown. A historical backlog review was performed of selected OE documents to determine applic   to PBAPS and adequacy of the review. In addition, both stations have adopted a near paper-less distribution process of operating experience. This facilitates a more efficient and timely distribution of operating experience. Another major improvement is associated with feedback to management. The status of this program open items is conveyed to the appropriate supervision and management via the use of self assessment activities and process performance indicators.

V. SUMMARY CONCLUSIONS

PECO Nuclear has effective processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence and reporting to the NRC. The PECO Nuclear problem identification and resolution process provides a appropriate threshold for problem identification, clear expectations for the identifiers, detailed operability and reportability determination guidance, root cause analysis techniques and assessment of generic implications to determine the extent of a problem, definition and implementation of corrective actions to prevent recurrence, and a focus on additional opportunities to improve. In addition, there are measures in-place to assure adequate training.

LR-C-10 (PEP) is the primary process for problem identification to management with an appropriately rigorous evaluation process (i.e., root cause analysis) and subsequent implementation of corrective actions to prevent recurrence. The LR-C-10 process invokes operability and reportability determinations to assure compliance with Technical Specifications and reporting requirements. The control of nonconformances by A-C-901 assures that hardware nonconformances found in installed plant equipment are identified, documented, evaluated, and resolved. A-C-901 also invokes operability and reportability determinations. Like LR-C-10, use of A-C-901 and the many other PECO Nuclear problem identification and/or corrective action processes, including the Quality Concerns Hotline, are part of management expectations for continuous improvement. The corrective action processes described herein provide reasonable assurance that if there are any issues with the design baseline documentation or processes for Limerick or Peach Bottom, these issues are identified and corrected through appropriate processes, including event investigation and corrective actions to prevent recurrence when warranted.

The operability and reportability processes are adequate to assure compliance with the NRC requirements. They are properly focused and provide a structured and documented review methodology.

The communications with the NRC Resident inspectors help to provide the Residents with sufficient information to ascertain whether PECO Nuclear 's facilities are operated safely and in compliance with our Facility Operating Licenses and regulatory requirements and commitments. Additionally, this interaction helps the residents to ascertain whether PECO Nuclear management programs are generally effective to assure both the safe design and operation of our facilities, and the health and safety of the public.

Effective process controls are in place to assure that changes to regulations, guidance or generic industry information are adequately considered and incorporated into the existing PECO Nuclear procedures or programs as appropriate. Necessary tools are provided to the users such that the tasks can be accomplished. Process training in-conjunction with specific task qualifications is also in place. In addition, the enhancements over the past few years have been considered a significant positive attribute to the success of the OEAP program. Lastly, the processes utilized to maintain management awareness of the program status were also evaluated and means are in place to keep management aware of adverse trends or problems.

There are many corrective action processes/procedures in place at PECO Nuclear. The principal program for problem identification, implementation of corrective actions, and actions to prevent recurrence is the Performance Enhancement Program (PEP). This program is complemented by the process for Control of Nonconformances (NCRs). These processes provide for consistent and well structured operability and reportability reviews to meet the NRC requirements, and industry and management expectations. In addition, PECO Nuclear has a process in place to routinely review generic NRC guidance, industry events, NRC research initiatives and industry initiatives. These programs provide a comprehensive process to evaluate both internal and external problems and events. These controls have been adequate in meeting NRC requirements and industry expectations.

ATTACHMENT 1

REQUEST (e)

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

RESPONSE

In order to provide a more cohesive response, information regarding the effectiveness of each of the requests (a, b, c and d) is provided within the response to the specific request. While this response to request e) summarizes the conclusions reached in each of the other responses (a,b,c and d), in order to reduce redundancy, it does not repeat the supporting information contained in the specific request responses.

PECO Nuclear has specific management expectations and assessment processes that provide for the direction and evaluation of the design bases processes and programs. PECO Nuclear's Policy Statement and Directive on Configuration Management indicate that configuration management of its nuclear facilities shall meet or exceed industry standards and good practices. This is accomplished through the implementation of integrated processes and procedures that identify existing plant design requirements and that control and document changes such that:

- selected plant structures, systems, components, and computer software conform to the approved design requirements,
- the plant's physical and functional characteristics are accurately reflected in selected plant documents in a timely mannor, and
- the configuration status is readily accessible to all organizations.

The policy is implemented by establishing and maintaining:

- documentation accurately reflecting the as-built condition of the plant,
- design basis information readily available and easily retrievable,
- plant structures, systems, and components (SSC) conforming to design basis requirements,
- integrated work processes that ensure changes in any functional area are implemented across all functional areas,
- processes that identify, control, and document changes to the design basis,
- a change control process that addresses justification, design, safety, implementation, maintenance, testing, operation, and verification of all plant modifications,
- self assessment mechanisms, status accounting and performance indicators providing feedback on the performance of the program,
- an accurate licensing basis,
- maintenance program activities which could affect the plant configuration, and
- training programs supporting configuration management.

The Directive categorizes the above Configuration Management processes and procedures into the following elements: scope evaluation, design requirements control, design changes control, and document management. The Directive requires that these elements be embedded within functional area work processes, i.e., Configuration Management is implemented by establishing and maintaining these elements within functional area procedural documents. Functional areas include but are not limited to: design basis management, drawing/document control, licensing, design change and temporary plant alteration processes, etc. In addition, the functional areas have mechanisms to assure proper evaluations and actions are performed when revising procedural documents to maintain the elements of Configuration Management. Also, the effectiveness of Configuration Management is evaluated on an ongoing basis by performing assessments and monitoring established Configuration Management related Performance Indicators (PIs).

I. ASSESSMENT PROCESSES

A. Internal PECO Nuclear Assessments

1. Self Assessments

PECO Nuclear performs periodic self assessments at LGS, PBAPS and Nuclear Group Headquarters. They are conducted utilizing the guidance of AG-CG-19, "Self Assessment Guideline." Self assessments are typically conducted at all levels of an organization. The assessments include a review of activities which an organization performs, for example, Engineering Change Request (ECR) quality, programs, procedure change processes and assessment processes. The assessment criteria entails identification of the strengths and weaknesses of the activity. This is accomplished by analyzing the activity or product in terms of the following:

- compliance with regulatory and procedure requirements,
- identification of safety related products,
- conclusions of independent assessments,
- effectiveness of previous corrective actions,
- customer feedback, and
- overall quality of product.

In addition to the periodic self assessments, there are three other types of self assessments performed at PECO Nuclear . They include; continuous, preemptive and reactive self assessments. Continuous self assessment is conducted on an ongoing basis. It promotes all work being performed correctly the first time. Preemptive or proactive self assessments are performed prior to a significant change. The purpose of this type of assessment is to identify and address potential problems before they occur. A reactive self assessment occurs when an independent assessment of a particular product identifies a problem that was not self identified. The reactive self assessment focuses on determining why the product owner did not self identify the problem and why the problem occurred.

Results of the self assessment activities are documented and subsequent action items are established and assigned as required. For situations where a condition adverse to quality is identified, Performance Enhancement Program (PEP) issues are initiated per LR-C-10 to obtain the appropriate evaluation of the situation and to determine corrective actions. Corrective actions that are developed to resolve an issue or close a performance gap are tracked to closure. Results of the self assessment activities have included procedure and process revisions and enhancements to work practices.

2. Engineering Assurance Assessments

Engineering Assurance (EA) within the Nuclear Engineering Division, is responsible for performing independent assessments of the programs, processes, modifications and work products impacting the stations and, specifically, the design bases of the plants.

These assessments are performed utilizing a technical methodology designed to improve the quality of our programs, processes and modifications. This assessment methodology incorporates three major review areas; technical, regulatory and business efficiency. When an assessment is performed, specific criteria is developed. This is to assure that the uniqueness of every product is assessed. To assist in the development of assessment criteria, NRC and industry documentation is also reviewed. Assessments employ a peer group of assessors selected by management. Recommended assessor performance and behaviors are provided.

The Quality Achievement Board Process focuses specifically on the modification work products. This process assesses the technical quality and configuration management integrity of work products prepared by the Site Engineering, Nuclear Engineering Division (NED) and contractors. The technical reviews are conducted utilizing technical subject matter experts (SMEs), experienced Lead Responsible Engineers (LREs) and program owners. There is a cadre of engineers and designers that have been selected by management to participate on these assessments.

3. Nuclear Quality Assurance

Nuclear Quality Assurance (NQA) oversight is accomplished using the Master Oversight Plan (MOP) which identifies areas to be evaluated, provides guidance for evaluation scope of each MOP area, identifies the evaluation frequency for each MOP area, identifies NQA responsibility for performing evaluations for each MOP area and identifies the documents w [illegible] provide the basis for the frequency of evaluations.

As defined in procedure NQA-39 "Control of the MOP", an "assessment" is a planned and documented activity performed in accordance with written procedures and checklists to determine or verify by observation, investigation, interviews, or evaluation of objective evidence the adequacy of, compliance with, and effectiveness of implementation of established procedures, instructions, drawings, and other applicable documents and elements of the quality assurance program.

In addition, as defined in procedure NQA-39 "Control of the MOP", a "surveillance" is an overview activity, other than an assessment, which may be used to fulfill the assessment process. Surveillances are also used to verify process acceptability and conformance to specified requirements, industry standards, and management expectations.

The current MOP identifies the topics, scope and frequency of NQA Assessments. Among the design and configuration control activity areas included in the MOP are Modification and Non-modification Engineering, Modification installation and testing activities, Temporary Plant Alterations, and Document and Vendor Manual Control. Assessments of these and other areas are utilized in part to evaluate various aspects of configuration management such as installation, testing, documentation, plant operation and configuration.

The NQA-21 procedure "NQA Assessments and Surveillances" describes the requirements for planning, scheduling, conducting and documenting NQA assessments and surveillances. The procedure is applicable to all NQA assessments and surveillances of internal activities for company-operated nuclear generating stations.

In addition to the formal scheduled assessments associated with the NQA MOP plan, the Independent Safety Engineering Groups (ISEG) at Limerick and Peach Bottom perform reviews with the primary objective to assess the overall quality of plant operations and to identify areas for improving plant nuclear safety. As part of these reviews, various configuration management activities are assessed by ISEG.

ATTACHMENT 1

B. Industry Evaluations

1. Joint Utility Management Audit (JUMA) Evaluations

To provide for an independent evaluation of selected NQA functions and products, an annual assessment is provided by a team of qualified audit personnel from other nuclear utilities. This independent review is called the Joint Utility Management Audit (JUMA). Collectively, the 1995 and 1996 JUMA reports covered NQA activities at Chesterbrook, Limerick and Peach Bottom. The reports identified that the NQA assessment program was being effectively conducted, that NQA was providing effective oversight of site engineering activities, and that assessment personnel met qualification requirements. No conditions adverse to quality were identified. The reports did contain actual and potential weaknesses. Additionally, recommendations for further improvement were offered for consideration. All JUMA identified items are evaluated by NQA personnel for action and are tracked to closure. JUMA reports are distributed to senior management, NQA management and the Nuclear Review Board.

2. Other Industry Evaluations

There are various industry organizations that perform both corporate and plant peer evaluations. These evaluations are conducted on a periodic basis to review performance in areas related to nuclear plant safety and reliability. These evaluations are performed based on a specified set of performance objectives and criteria. The performance objectives are broad in scope and generally cover single, well defined management areas. The supporting criteria are more narrow in scope and typically describe specific activities that contribute to the achievement of the performance objectives.

The evaluations focus on such areas as configuration management; design control; plant modifications; engineering procedures and documentation; document control; technical support; operating experience review; training and qualification; and independent monitoring and assessment. These evaluations help to ensure: an effective configuration management process is in place; the plant design is properly implemented; plant characteristics are accurately reflected in plant documents; design documentation is current and readily available; in-house and industry operating experience is evaluated and actions are taken to improve plant reliability and safety, as appropriate; engineering personnel are trained, qualified and knowledgeable of ongoing changes to the plant; and periodic reviews are performed to assess the effectiveness of the programs associated with configuration management and design control.

C. NRC Evaluations

The NRC periodically reviews the programs that revise the design bases through the reactor inspection program. The objective of the NRC's inspection program is to obtain sufficient information, primarily through direct observation and verification of licensee activities, to ascertain whether reactor facilities are operated safely and in compliance with license and regulatory requirements. Additionally, this program attempts to ascertain whether licensee management programs are effectively implemented to ensure both safe design, construction, and operation of reactor facilities and the health and safety of the public. The intent of the inspection program is to provide the desired level of assurance that licensees are complying with NRC regulations, rules, orders, and license provisions, and are taking appropriate actions to protect nuclear materials and facilities, the environment, and the health and safety of the public.

Inspections provide input to the Systematic Assessment of Licensee Performance (SALP) program evaluations. The objectives of the SALP process are: to conduct an integrated assessment of licensee safety performance that focuses on the safety significance of NRC findings and conclusions during the assessment period, and to provide a vehicle for meaningful dialogue with the licensee regarding its safety performance based on the insights gained from synthesis of the NRC observations. Four functional areas are evaluated during the SALP process, including Operations, Maintenance, Engineering, and Plant Support. As a result, SALP is an important and integral part of the overall NRC inspection program and evaluation process. Overall, the various NRC inspection programs and review processes emphasize achieving a balanced look at a cross section of licensee activities important to plant safety and reliability.

D. Assessment Processes Conclusions

Effective oversight and assessment is accomplished using the combination of PECO Nuclear organizational assessments (self assessments and oversight assessments ,e.g., engineering assurance assessments), PECO Nuclear independent internal oversight (Nuclear Quality Assurance (NQA) using the Master Oversight Plan (MOP) and ISEG assessments), PECO Energy Nuclear Review Board (NRB), and Industry peer evaluations. These assessments help to assure the adequacy of configuration management (CM) and design control, and the availability of design bases information. The results of these evaluations identify strengths in addition to areas for improvement. Addressing the areas in need of improvement assists PECO Nuclear in ongoing efforts to improve all aspects of its nuclear programs.

In addition, the NRC inspection program emphasizes achieving a balanced look at a cross section of licensee activities important to plant safety and reliability. NRC inspectors perform a basic mission in determining that a licensee operates the plant safely and meets current regulatory requirements and commitments. The NRC inspection activities and other performance reviews ensure that PECO Nuclear 's engineering activities maintain good design control, configuration management, and the availability of its design-bases information.

II. SUMMARY OF EFFECTIVENESS AND CONCLUSIONS

PECO Nuclear has performed an extensive assessment in order to provide a complete and thorough response to this request for information. This assessment reviewed the current CM processes for design control, procedure control, SSC performance, and Problem Identification/Corrective Action; reviewed the efforts which PECO Nuclear has proactively performed in response to various industry events, industry initiatives and NRC identified issues; and reviewed the extensive internal and external assessment processes and their associated results.

The results of this assessment indicate that PECO Nuclear has been proactive in response to generic regulatory and industry concerns and initiatives regarding effective control of the design bases and that there is reasonable assurance that the configuration management processes in place are adequate and effective in controlling the configuration of the design bases of Limerick Generating Station and Peach Bottom Atomic Power Station.

PECO Nuclear has reasonable assurance that engineering design and configuration control mechanisms are in place to ensure that the design bases are maintained. These mechanisms are an integration of procedural, programmatic, and management controls which ensure that the plants conform to approved design requirements, the physical and functional characteristics of the plants are accurately reflected in controlled documents, the status of design, plant, and functional design changes and temporary plant alterations and associated documents are readily accessible to appropriate line organizations and the licensing bases are accurately maintained. However, in reaching this conclusion as part of the request (a) response, the assessments indicated that the area of software quality needs improvement. Specifically, problems were encountered in certain changes to PIMS software due to incomplete design requirements and inadequate testing of software interfaces. In the plant process computer area, some problems have been related to incorrectly configured database elements which resulted in NSSS software modules halting. In both of these areas, PEP issues were generated to identify root causes and corrective actions have been initiated. In mid-1996, in order to improve software quality, PECO Nuclear initiated an in-depth review of current software management practices. In support of this initiative, a review was recently completed by an independent consultant which concluded that improvements are needed in the area of software design, interfaces, testing and data control. The assessment of the recognized weakness of software configuration control will be completed and appropriate corrective actions implemented. PECO Nuclear will provide additional information in future correspondence (see commitment 5 in Attachment 3). The need for this corrective action does not change the conclusion that there is reasonable assurance that engineering design and configuration control mechanisms are in place to ensure that the design bases are maintained.

Based on the projects, assessments, and corrective actions, PECO Nuclear concludes there is reasonable assurance that the current PBAPS and LGS maintenance, operating, and testing procedures adequately reflect the station design bases. A review of the procedure control process indicates the process appropriately addresses design bases requirements and has been adequately implemented. In addition, a review of the processes for changing the design bases indicates that the processes have the procedural mechanisms to identify the impacted station procedures and revise them. However, in reaching this conclusion as part of the request (b) response, the assessments indicated that, although the procedural mechanisms are being executed, the implementation of the process requires strengthening to ensure all impacted procedures are updated as required. An assessment of the recognized weakness to identify and revise procedures impacted by design basis changes will be performed and appropriate corrective actions implemented. PECO Nuclear will provide additional information in future correspondence. Closure of identified discrepancies continues. Corrective actions associated with the program concerns will be identified and tracked via the PEP process (see commitment 4 in Attachment 3). The number of procedures that require revisions due to design changes is small compared to the body of procedures that implement design bases requirements, and the processes for updating procedures due to design changes are generally being followed. A review of the 1996 procedure revision cause code found that the missed procedure revisions had minimal impact on plant safety. Therefore, there is reasonable assurance that procedures adequately reflect the station design bases. The need for this corrective action does not change the conclusion that there is reasonable assurance that the current PBAPS and LGS maintenance, operating, and testing procedures adequately reflect the station design bases.

PECO Nuclear has determined that there is reasonable assurance that the configuration and performance of the Peach Bottom and Limerick station's systems, structures and components (SSCs) are consistent with the design bases. This determination is based on an extensive assessment of past and existing PECO Nuclear programs and processes where design bases review was the primary purpose, plus other programs or tasks where design bases verification was a benefit of the activity. The programs reviewed generally included physical walkdowns to verify the SSC versus design bases information. There have been several comprehensive programs which have compared existing plant configuration with its associated design bases documentation. The multiple PECO and NRC SSFIs, and the recent UFSAR Verification Program at both stations were broad scope activities which verified configuration consistency. The power rerate project and heat load/flow modeling of the cooling water systems regeneration project both verified and regenerated the affected systems design bases via plant modifications and calculation revisions. The other programs and processes identified in this summary have independently verified the configuration and performance of specific systems, components or attributes. Cumulatively, these activities provide reasonable assurance that the overall SSC configuration and performance are consistent with the plant design bases. In reaching this conclusion as part of the request (c) and Attachment 2 responses, it was identified that the UFSAR verification process identified prudent enhancements (see commitment 1 in Attachment 3) and that the DBD maintenance effectiveness review has identified a weakness in the technical resolution of DBD open items and DBD

ATTACHMENT 1

maintenance activities (see commitment 3 in Attachment 3). Corrective actions associated with the program concerns will be identified and tracked via the PEP process. The need for these corrective actions does not change the conclusion that there is reasonable assurance that the overall SSC configuration and performance is consistent with the plant design bases.

The combination of effective problem identification and corrective action processes, an appropriate operability and reportability decision process, and a structured external event evaluation process supports the continuous improvement culture at PECO Nuclear and provides assurance that problems are identified and resolved.

Overall, PECO Nuclear has been proactive in response to generic regulatory and industry concerns and initiatives regarding effective control of the design bases. Issues for improving PECO Nuclear performance have been identified with commitments for appropriate problem definition and corrective actions in Attachment 3. In addition, as outlined in commitment 2 in Attachment 3, PECO Nuclear is committed to assessment of Configuration Management performance and will perform two (2) SSFI type inspections in 1997. Performing SSFIs on a sample of critical systems is an accepted and sound approach to validating the design and implementation processes. Observations and other opportunities for improved performance have also been identified and are being evaluated by PECO Nuclear management. Nonetheless, there is reasonable assurance that the configuration management processes in place are adequate and effective in controlling the configuration of the design bases of Limerick Generating Station and Peach Bottom Atomic Power Station.

ATTACHMENT 2

REQUEST

Indicate whether you have undertaken any design review or reconstitution programs, and if not, a rationale for not implementing such a program.

RESPONSE

The plant design bases for Peach Bottom Atomic Power Station (PBAPS) and Limerick Generating Station (LGS) have changed from the time of their initial operating licenses. These changes have occurred as a result of plant configuration modifications which were initiated for economic, performance, and regulatory reasons. PECO Nuclear has conducted comprehensive programs and established continuing processes to provide reasonable assurance that the plant design baseline is: (1) as stated in existing documentation, (2) accessible for use by PECO Nuclear personnel, and (3) maintained current through the use of configuration control programs and procedures. PECO Nuclear has performed an extensive assessment focused on the quality and completeness of these programs and processes to confirm that these three attributes remain true for the PECO Nuclear stations.

This review focused on programs or processes where design bases review was the primary purpose, plus programs or tasks where limited design bases verification was a benefit of an activity with a different goal. Each program was assessed for its value in verifying, regenerating and/or disseminating design bases information. Findings and corrective actions were evaluated for their impact on the objective of the specific activity and overall goal of design bases control.

Several programs for which the primary purpose was to verify and/or regenerate design bases information were identified by the review. These programs provide a basis for concluding that the design bases are intact. The Design Baseline Document effort was the most comprehensive project to capture and organize design bases information. This effort combined design bases information from diverse sources into a useful reference document which is readily available for plant verification or modification activities.

The UFSAR Verification effort, performed at both stations from May through September 1996, was a proactive response to industry events and revealed inconsistencies between UFSAR sections and design bases information. None of the discrepancies identified during this review resulted in an Unreviewed Safety Question (USQ).

Additional programs which were evaluated as part of this review included Improved Technical Specifications, Power Rerate, Setpoint Control, Commitment Annotation, Commitment Tracking Historical Backlog, Component Record List, and the Integrated Nuclear Data Management Systems.

I. DESCRIPTION OF DESIGN BASES REVIEW:

The following identifies specific design bases verification or regeneration efforts which have been performed by PECO Nuclear during the life of the Peach Bottom and Limerick stations. Unless otherwise noted, the programs described and discussed were/are common to both stations.

A. Design Baseline Document Project

The Design Baseline Document (DBD) project was initiated in 1988 concurrent with the Peach Bottom restart effort. The objective of the DBD project was to identify the design bases requirements and the implementing design baseline which fulfilled those requirements, locate or redevelop missing design information as needed, and present the information in a consistent format available to all plant support personnel.

ATTACHMENT 2

This project responded to industry initiatives at the time to consolidate and validate design bases information such that it could be readily available to assist plant operations and engineering personnel. The project also responded to an INPO assessment of configuration management practices which identified that design bases information was not readily available to support design and inspection activities. The objective of the project was to provide a single-source road map to design bases information to streamline engineering and operating activities where knowledge and/or understanding of the underlying bases or "whys" of system and topical subject area design was required. The DBD project, in conjunction with other configuration management activities initiated at the time, was implemented to make design and design basis information easier to identify and readily accessible.

The project was conducted in four major activities: information indexing, Design Baseline Document (DBD) preparation, DBD technical open item resolution, and establishment of a DBD maintenance process.

The project was based on a thorough review of design bases reconstitution programs being implemented by other utilities, recommendations provided by NUMARC (in their draft of NUMARC 90-12) and other industry groups, and internal surveys to identify user needs. The objectives and the level of detail for the DBDs were established based on these reviews and surveys. As a result of these surveys, PECO Nuclear DBDs were designed to focus on the upper tier system design bases requirements, which remain relatively constant over plant life, and provide references to the design details. Design information which is controlled under the design control process was not generally reiterated in the DBDs. Instead, the DBDs reference the controlling documents which support the design information presented in the DBD to minimize duplication and reduce the risk of document update inconsistencies. PECO Nuclear's DBD program was compared to the drafts of NUMARC 90-12 and found to be consistent with the NUMARC guidance with only minor differences.

Prior to DBD preparation, an extensive design information cataloguing effort was performed to identify all sources of controlled documentation and records and to associate them with the planned DBDs. During DBD development, a sampling review of calculations confirmed the program assumption that existing calculations and analyses could be used to identify the design baseline without performing a complete validation of each analysis.

All safety-related plant systems, most balance-of-plant systems, and several topical subject areas were addressed. One hundred fifty (150) DBDs were originally planned. As DBD production progressed, the list of planned DBDs was continually examined in light of lessons learned during early DBD preparation. Some systems and topical subjects were removed from the planned list, the scopes of some DBDs were increased, and other DBDs were split into multiple DBDs or combined to better present the information. When the project was completed, 152 DBDs were prepared, consisting of 62 system DBDs and 13 topical DBDs for each plant plus 2 topical DBDs which are common between the plants. The DBDs were prepared by reviewing sources of design and licensing requirements and commitments to develop a consolidated list of requirements applicable to plant systems, structures, and components. Using the list of design bases requirements as a guide, current as-built design documentation was reviewed to identify the design functions and features which fulfilled the design requirements.

DBD preparation relied on existing design documentation to identify the plant configuration and did not attempt to verify that the design documentation was consistent with the as-built plant through project specific walk-downs. Discrepancies between design documentation and the physical plant were/are identified and corrected as part of continual work processes and periodic inspections performed for other activities. During initial DBD preparation, inconsistencies between design bases requirements and design documentation or instances of missing information were identified and resolved as part of the DBD open items process.

DBD preparation activities included a review to verify that all system design bases controlling parameters as identified in the DBDs were being tested in the plants. Approximately 20 to 30 station procedures per system were reviewed to verify that the DBD controlling parameters were supported by the system test acceptance criteria.

Once prepared and issued, the DBDs were entered into the document control system as controlled documents. Design control procedures were revised to integrate the DBDs into the design change review process to maintain the DBDs consistent with evolving plant design or to create new DBDs as needed.

Maintaining design bases information current is enhanced by the ability to easily identify design bases information and by incorporating specific verifications into the design change process. As the initial DBD preparation project was completed, the DBDs were assigned engineering branch owners who are responsible for ensuring that the documents are adequately maintained.

Assessments of the DBD maintenance process have been performed annually since 1994, and have indicated that there is room for improvement in the administrative and technical areas of the DBD maintenance process. As a result, the process controlling DBD updates was revised to streamline the requirements governing DBD maintenance and strengthen the requirements governing DBDs in the design change process. The DBD maintenance process is currently being re-evaluated to determine its efficiency and effectiveness. Results to date indicate that there are opportunities to further improve the DBD maintenance process. The majority of the items identified by the assessment pertain to DBD format and information consistency. Another area being evaluated is the technical adequacy of changes or additions being made to DBDs. A sampling of ECRs which affect DBDs is being evaluated for technical accuracy and adequacy. Appropriate corrective actions will be initiated based on the evaluation (see commitment 3 in Attachment 3).

Design bases information is made readily accessible via the DBDs. These documents provide a single-source mapping to design bases information via an extensive cross referencing scheme within the DBDs between source documents and the information presented in the DBDs.

These efforts to reconcile design bases and design configuration documentation have resulted in a consistent, current, and available presentation of design baseline information. The processes are designed with continuous improvement through use of mechanisms to continually assess and improve DBDs.

B. UFSAR Verification Program

The purpose of the UFSAR Verification effort was to develop and implement a methodology for verifying and remedying the accuracy of selected information contained in the Limerick and Peach Bottom UFSARs. The review of the UFSAR documents was formally implemented, at both Stations, from May through September 1996. This review program was a proactive response by PECO Nuclear Executive Management to recent industry events.

The scope of the UFSAR verification program included an original scope of 30 UFSAR sections for each station. The original selection criteria for sections reviewed included Probabilistic Safety Assessment (PSA) significant items, sections with high change activity and random sections. The scope was expanded during the review process, at the direction of the section review leads, to include an additional 53 sections at PBAPS and an additional 68 sections at LGS, resulting in a total review of approximately 20% of each UFSAR. The total review process and corrective action follow-up involved an estimated 4 man-years per UFSAR reviewed. The program scope included identification and resolution of identified inconsistencies with the document and the as-built/as-found facility.

The program was implemented at each station, to review and verify the accuracy of selected information contained in the Station's UFSAR. This program involved the following review activities:

ATTACHMENT 2

(1) each team was assigned a section to review (each section was selected as describing a plant system), (2) section review expanded, in most cases, to encompass additional sections referenced from within the original section, (3) inconsistencies were categorized as "typos", "incorrect statements" and/or "ambiguous statements" and assigned a significance level based upon safety and/or regulatory impact, (4) management review panel for concurrence of review depth and proper characterization of any findings, and (5) a PECO Nuclear Engineering Council review. The management review panel consisted of station management, engineering, operations and licensing personnel.

The review identified about 450 discrepancies at Limerick and about 600 at Peach Bottom. The largest percentage of these discrepancies included statements within the UFSAR which were categorized as "ambiguous", i.e., information contained within the UFSAR which could be misinterpreted. "Incorrect" statements amounted to approximately 9% (at Peach Bottom) and 16% (at Limerick) of the identified discrepancies. However, a small number of incorrect discrepancies (5 at LGS, 6 at PBAPS) were determined to be nonconforming conditions. No incorrect statements identified resulted in safety significant issues or an Unreviewed Safety Question (USQ).

Recommendations for process enhancements were provided to the PECO Nuclear Engineering Council following the program team's initial assessment. Recommendations included implementing a guideline for administrative control and maintenance of the Personal Librarian Software (PLS) database and overall changes to the ownership of the UFSAR sections. The use of the PLS tool, which facilitates SAR reviews for 10CFR50.59 Reviews, should be communicated and demonstrated to all personnel involved with the 10CFR50.59 process. Training for organizations outside of the engineering organization and to new engineering personnel should be focused to include sensitization to the information contained within the UFSAR and overall UFSAR fidelity. This training would be incorporated into station continuing training lesson plans. A PEP issue was generated to evaluate and identify any Conditions Adverse to Quality (CAQ), root causes, generic implications and corrective actions through a root cause analysis. The PEP process will also track the previously listed recommendations to closure.

Due to the number of discrepancies identified, a few remain tracked as open items within the PECO Nuclear corrective action system. All nonconforming conditions identified during this review have been dispositioned via common procedures A-C-901 and MOD-C-9. All other discrepancies identified, with regard to the "incorrect," "ambiguous," or "typos" are being evaluated and resolved, in an expedited manner, via the Engineering Change Request (ECR) process (parent procedure MOD-C-9). Changes to the UFSAR are undergoing a review per the criteria in common procedure LR-C-13 and a 10CFR50.59 Review, as applicable.

PECO Nuclear is evaluating the scope and schedule of the completion of the UFSAR verification and will provide additional information in future correspondence (see commitment 1 in Attachment 3). Closure of identified discrepancies continues. Corrective actions associated with the program concerns will be identified and tracked via the PEP process once the review is completed. Additional baseline and continuing training of nuclear personnel will provide continued focus on UFSAR content, accuracy and fidelity.

C. Component Record List (CRL)

The CRL originated as a configuration management effort to consolidate component information which, at the time, was contained in several different lists and databases such as the Q List, fuse index, and Equipment Qualification list. Following the initial data acquisition effort, the CRL has evolved to provide increasing functionality as needs were identified. Presently, the CRL defines component IDs for the plants, includes the Q List for PBAPS and LGS, Codes and Classifications information, Nuclear Plant Reliability Data System (NPRDS) data, equipment qualification information, and numerous other data on plant components and contains component design bases for the plant. Component labeling and nomenclature is almost entirely based on the information in the CRL; this referencing standard is used throughout PECO Nuclear processes.

Verification activities for the CRL primarily occurred during data loading of several previously existing data bases and did not normally include field verification activities. Field verification activities were typically performed during resolution of problems identified aside from the CRL data loading activities.

The CRL plays a critical role in plant design and operation. Both the CRL data and the CRL maintenance process are components of PIMS which are available to all plant personnel. The CRL maintenance or change process is integrated with the design change process via a PIMS electronic change mechanism providing automated CRL change incorporation when changes are implemented in the plant. Numerous plant activities are PIMS based, relying on the CRL to provide comprehensive component data on components. Routine activities, such as maintenance, testing, and procurement, interface with the CRL and make it easier to identify discrepancies between CRL data and the physical plant. When CRL discrepancies are identified, they are resolved using appropriate corrective action processes. Each of these processes incorporates a "generic implication" review to determine if an identified issue is a random occurrence or an indicator of a bigger concern. Additionally, several internal self assessments have been performed on the CRL during its existence. Each focused on particular CRL data and identified and corrected discrepancies. As a result, the CRL has been reviewed from several different perspectives, further improving data quality. The general approach of corrective action processes and self assessments provides a continuously improving mechanism to increase the rationale that data in the CRL is consistent with the physical plant.

D. Improved Technical Specifications (Peach Bottom Only)

Improved Technical Specifications (ITS) were implemented at Peach Bottom by replacing the original specifications in their entirety with a version consistent with NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4."

Generation of the PBAPS ITS involved an extensive interdisciplinary review to ensure that they accurately reflected the plant, the UFSAR, and other pertinent documentation. The plant design characteristics, existing Technical Specifications, and Standard Technical Specifications were compared to selection criteria developed using NUREG-1433. In some cases, plant tests were performed to verify that proposed specifications and testing was appropriate. Actual surveillance history of plant equipment was an input into new surveillance test frequencies.

The entire Technical Specifications were validated against the design bases during ITS development. Detailed bases are now located with the Technical Specifications. Use of the ITS enables better configuration management in accordance with design and license considerations because the new specifications include fully developed bases that clearly define requirements for operability.

E. Power Rerate Program

The Power Rerate Program scope included all necessary activities to assure safe, reliable operation at 105% power and included a comprehensive review of plant design bases. Both systems operation and current design bases were thoroughly reviewed for impact of 105% power operations. The rerate program began in 1992 with final acceptance testing at rerated conditions completed during Unit refueling outages in 1994, 1995, and 1996.

ATTACHMENT 2

Power Rerate involved a comprehensive look at plant design bases to determine if the existing controlling parameters were consistent with the plant and able to remain within design bases at 105% power. Extensive System Manager involvement in the program assured that the engineering data, including the rerate inputs and performance data and the calculation outputs matched their knowledge of the system operation and system configuration. Current system performance data was used to assure any degradation of equipment over plant life and any replacement equipment was incorporated into the analysis. Comprehensive design review and recalculation of calculations was performed to assure that the power plant would operate per design at 105% power. In the vast majority of areas, the plant and design bases were consistent and calculations were either recalculated at 105% power conditions or reviewed to determine acceptability at 105% power conditions. Where required the calculations were recalculated or reconstituted. Appropriate testing and monitoring was performed to assure plant response at 105% was appropriate.

Significant testing was performed at each of the Units following implementation of Power Rerate. The System Managers, System Manager Review Board and PORC at PBAPS, Supervisors, and PORC at LGS performed a structured review of rerate results to assure that the results were consistent with the System Managers' and plant management's understanding of current plant configuration.

F. Improved Instrument Setpoint Control Program

The Improved Instrument Setpoint Control Program (IISCP) is a computerized calculation software which was implemented by PECO Nuclear to upgrade previous hand calculation methods. The IISCP software was developed based on commonality among users, current industry practices, regulatory requirements, configuration management, and other company programs that impact or are affected by a setpoint control program. The new software performs calculations to current standards using a consistent methodology; error terms (tolerances, environmental effects, etc.) are specifically identified. The identification and regeneration of in-scope design bases calculations were completed for approximately 1600 instrument channels using the IISCP software in 1994. These components included all Technical Specification instruments plus instruments affected by power rerate.

The IISCP software maintains instrument setpoints in accordance with the plant's design parameters by imposing a consistent method of determining setpoints and controlling the revision, review, and approval process. The program software is available on PECO Nuclear's wide area network so that setpoint data from either site can be viewed from many PECO Nuclear locations.

G. Historical Commitment Backlog Review Effort / Commitment Annotation Program

During the implementation of the Commitment Tracking Program in 1988, it was identified that some commitments issued prior to 1988 may not have been properly implemented in PECO Nuclear programs. Further, continued compliance with commitments involving NRC, INPO, and ANI could not be easily verified because prior to 1988, PECO Nuclear did not have a formal commitment tracking program.

The Historical Commitment Backlog Review Effort located and evaluated several thousand documents issued and received by PECO Nuclear from 1974 to 1988. The final screening resulted in the annotation of over 1,000 commitments in PECO Nuclear procedures/programs. No significant non-compliances with these historical commitments were identified by PECO Nuclear during the disposition phase of the review effort.

The scope of the Commitment Annotation Program includes annotation of implementing documents with programmatic commitments (i.e., ongoing actions) made by PECO Nuclear to external organizations, and annotation of implementing documents with significant programmatic corrective actions associated with the review of industry operating experience (OEAP) and significant PEP issues. The Commitment Annotation Program provides assurance that commitments/corrective actions implemented by PECO Nuclear are properly maintained.

In 1992 and 1993, as a separate task of the Historical Commitment Backlog Review Effort, Limerick performed a review of selected UFSAR commitments to verify the commitments were incorporated in Station procedures. Sections of the LGS UFSAR which contained specific commitments regarding plant operation or specific commitments to a document, such as a Regulatory Guide or Industry Standard, were identified. Examples of commitments regarding plant operation include: inspection of systems during normal operation to ensure minimal leakage, or requirements for maintaining valves closed by administrative means during certain plant conditions. Once identified, these UFSAR commitments were reviewed for accuracy and station procedures were reviewed to ensure the item was incorporated. Approximately 400 UFSAR commitments were verified to be implemented in station procedures.

The Commitment Annotation Program provides assurance that commitments/corrective actions implemented by PECO Nuclear are properly maintained. The program provides value by ensuring that personnel do not unknowingly remove programmatic commitments/corrective actions from procedures/ programs and thus expose PECO Nuclear facilities to non-compliance with external regulations and requirements or to the potential for repeat events at the stations. In addition, at Limerick approximately 400 UFSAR commitments were verified to be implemented in station procedures.

H. Integrated Nuclear Data Management System

The Integrated Nuclear Data Management System (INDMS) is a design quality computer program that provides a comprehensive cable management system for Peach Bottom and Limerick. INDMS is used to accomplish the following specific functions:

- Used for designing, maintaining, and controlling cable, circuit, raceway, and junction box information,
- Integrates various fire protection and safe shutdown, analysis capabilities using comprehensive cable data,
- Tracks both the as-designed and as-built plant configuration,
- Maintains cable design information, such as cable calculations, for the review of raceway fill, cable tray overfill, ampacity derating, and cable combustible loading by room.

During initial data input to the INDMS database in 1992, many data reconciliation processes were performed to identify discrepant information. These processes included physical walkdowns of cable routings to verify as-built conditions. The plant design baseline for cabling is developed, maintained current, documented, and accessible via this controlled computer program.

II. SUMMARY CONCLUSION:

PECO Nuclear has reasonable assurance that programs and processes are in place to ensure the design bases for Peach Bottom and Limerick are (1) as stated in existing documentation, (2) accessible for use by PECO Nuclear personnel, and (3) maintained current through the use of configuration control programs and procedures. This conclusion is based on an extensive assessment of past and existing PECO Nuclear programs and processes where design bases review was the primary purpose, plus other programs or tasks where design bases verification was a benefit of the activity. The response to request

ATTACHMENT 2

(a) contains a description of the applicable processes used by PECO Nuclear to control and maintain the design bases.

Programs, such as the Design Baseline Document Project and the recent UFSAR Verification effort, were conducted to compile and verify design bases information. The results of these and the other specific programs discussed confirm that the design bases contained in the UFSAR are reflected in the DBDs, design drawings, controlled data listings, and other design bases documentation. Procedural controls are in place to assure that the design bases information is maintained current by recognizing and controlling the many sources of bases changes.

PECO Nuclear performs routine self-assessments of the above programs to assure their effectiveness. Configuration management performance indicators identify trends in program performance which permits timely corrective action when necessary.

The DBD maintenance effectiveness review has identified weaknesses in the technical resolution of DBD open items and DBD maintenance activities. These issues are being evaluated and appropriate corrective actions will be implemented (see commitment 3 in Attachment 3).

ATTACHMENT 3

Commitments made as a result of this response:

1. Completion of Updated Final Safety Analysis Report (UFSAR) Verification

ISSUE:

A review of the UFSAR documents was formally implemented, at both Stations, from May through September 1996. This review program was a proactive response by PECO Nuclear Senior Management to recent industry events. The scope of the UFSAR verification program included an original scope of 30 UFSAR sections for each station. The original selection criteria for sections reviewed included PSA significant items, sections with high change activity and random sections. The scope was expanded during the review process, at the direction of the section review leads, to include an additional 53 sections at PBAPS and an additional 68 sections at LGS, resulting in a total review of approximately 20% of each UFSAR.

The review identified about 450 discrepancies at Limerick and about 600 at Peach Bottom. The largest percentage of these discrepancies included statements within the UFSAR which were categorized as "ambiguous", i.e., information contained within the UFSAR which could be misinterpreted. "Incorrect" statements accounted for approximately 9% (at Peach Bottom) and 16% (at Limerick) of the identified discrepancies. However, a small number of incorrect discrepancies (5 at LGS, 6 at PBAPS) were determined to be nonconforming conditions. No incorrect statements identified resulted in safety significant issues or an Unreviewed Safety Question (USQ).

Recommendations for process enhancements were provided to the Nuclear Engineering Council following the program team's initial assessment. A PEP issue was generated to evaluate and identify any Conditions Adverse to Quality (CAQ), root causes, generic implications and corrective actions through a root cause analysis. The PEP process will also track the recommendations to closure.

COMMITMENT:

PECO Nuclear is committed to complete the verification of the UFSAR and is evaluating the scope and schedule of the completion of the UFSAR verification project. PECO Nuclear will provide additional information on the scope and schedule of the completion of the UFSAR verification project within 90 days of the date of this letter.

2. Completion of two Safety System Functional Inspections (SSFIs) during 1997

ISSUE:

Safety System Functional Inspections (SSFIs) were conducted on a sample of systems from 1990 to 1994. The objectives of the SSFIs were to determine if the systems as designed, installed and configured were capable of performing their intended safety functions and to determine the effectiveness of engineering and technical support processes as they relate to the systems' ability to perform their intended safety functions. SSFIs were conducted by PECO Nuclear at LGS on High Pressure Coolant Injection, Feedwater, Emergency Service Water, and Diesel Generators and AC Electrical Distribution systems. SSFIs were conducted by PECO Nuclear at PBAPS on 125/250 VDC Distribution, High Pressure Service Water, Diesel Generator and AC Emergency Electrical Distribution, Stand-by Gas Treatment, Compressed Air and Feedwater Systems. SSFIs were conducted by the NRC at PBAPS on the Emergency Service Water System and High Pressure Coolant Injection System and on both the LGS and PBAPS

ATTACHMENT 3

Electrical Distribution Systems. The SSFI's employed a "deep vertical slice" methodology and team interaction techniques developed as detailed in NSAC 121, November 1988.

COMMITMENT:

Performing SSFI type inspections on a sample of critical systems or processes is an accepted and sound approach to validating the design and implementation processes. PECO Nuclear is committed to assessment of configuration management performance and will perform two (2) SSFI type inspections in 1997. The continuance of this program will be evaluated upon completion of the two (2) SSFI type inspections.

3. Completion of Corrective Actions associated with Design Baseline Document (DBD) maintenance

ISSUE:

Assessments of the DBD maintenance process have been performed annually since 1994, and have indicated that there is room for improvement in the administrative and technical areas of the DBD maintenance process. As a result, the process controlling DBD updates was revised to streamline the requirements governing DBD maintenance and strengthen the requirements governing DBDs in the design change process. The DBD maintenance process is currently being re-evaluated to determine its efficiency and effectiveness. Results to date indicate that there are opportunities to further improve the DBD maintenance process. The majority of the items identified by the assessment pertain to DBD format and information consistency. Another area being evaluated is the technical adequacy of changes or additions being made to DBDs. A sampling of ECRs which affect DBDs is being evaluated for technical accuracy and adequacy. Appropriate corrective actions will be initiated based on the evaluation.

COMMITMENT:

PECO Nuclear is committed to complete an evaluation of the process, procedure and division of responsibility for Design Baseline Document maintenance and is evaluating the scope and schedule of this task. Corrective actions associated with the program concerns will be identified and tracked via the PEP process. PECO Nuclear will provide written confirmation to the NRC within 30 days of completion of this evaluation and identification of corrective actions.

4. Completion of Corrective Actions associated with the identification of affected procedures as a result of design changes.

ISSUE:

A review of the processes for changing the design bases indicates that the processes have the procedural mechanisms to identify the impacted station procedures and revise them. However, numerous assessments have indicated that implementation of the process requires strengthening to assure all impacted procedures are updated as required.

A review of NRC Inspection Reports, PEP issues, NQA audits, Self Assessments, and ISEG Reports indicates a number of deficiencies have occurred in station procedures due to the inadequate incorporation of design changes into the procedures. Recent data has indicated that the corrective actions taken have not adequately resolved this problem.

The number of procedures that require revisions due to design changes is small compared to the

ATTACHMENT 3

body of procedures that implement design basis requirements and the processes for updating procedures due to design changes are generally being followed. A review of the 1996 procedure revision cause code data found that the missed procedure revisions had minimal impact on plant safety. Therefore, there is reasonable assurance that procedures adequately reflect the station design basis.

COMMITMENT:

An evaluation of the recognized weakness to identify and revise procedures impacted by design basis changes will be performed and appropriate corrective actions implemented. Closure of identified discrepancies continues. Corrective actions associated with the program concerns will be identified and tracked via the PEP process. PECO Nuclear will provide written confirmation to the NRC within 30 days of completion of this evaluation and identification of corrective actions.

5. Completion of Corrective Actions associated with software configuration control.

ISSUE:

A prior review of Performance Enhancement Program (PEP) issues and assessments revealed that the area of software quality needs improvement. Specifically, problems were encountered in certain changes to PIMS software due to incomplete design requirements and inadequate testing of software interfaces. In the plant process computer area, some problems have been related to incorrectly configured database elements which resulted in NSSS software modules halting. In both of these areas, PEP issues were generated to identify root causes and corrective actions have been initiated. In mid-1996, in order to improve software quality, PECO Nuclear initiated an in-depth review of current software management practices. In support of this initiative, a review was recently completed by an independent consultant which concluded that improvements are needed in the area of software design, interfaces, testing and data control. Recommendations were made to strengthen software quality assurance procedures and to adopt an industry accepted software development model in order to create a more rigorous software CM process. In addition, NQA and ISEG personnel have been actively involved in several assessments to ensure root causes are determined and corrective actions identified.

COMMITMENT:

The evaluation of the recognized weakness of software configuration control will be completed and appropriate corrective actions implemented. Corrective actions associated with the software configuration control concerns will be identified and tracked via the PEP process. PECO Nuclear will provide written confirmation to the NRC within 30 days of completion of this evaluation and identification of corrective actions.

ATTACHMENT 4

Glossary of Terms

10 CFR 50.59 Review	At PECO Nuclear this is a two stage process comprised of a **Determination** and **Safety Evaluation**: If the **Determination** concludes that a **Safety Evaluation** is not necessary for an activity, then the **Determination** constitutes the 10CFR50.59 Review for that activity. If the **Determination** concludes that at **Safety Evaluation** is required, then the **Determination** <u>and</u> **Safety Evaluation** constitute the 10CFR50.59 Review for that activity. The **Safety Evaluation** concludes whether or not an Unreviewed Safety Question (USQ) exists and prior NRC approval is required.
Common	Procedures, programs or processes which apply to more than one Department and usually to both sites.
Condition Adverse to Quality (CAQ)	A condition where procedures, work processes, or activities permit the potential for, contribute to, or result in failures, malfunctions, deficiencies, deviations, defective material or equipment, or noncompliance with specified requirements.
Design Bases	The term "design bases" as used in this request, is defined in the same manner as in 10 CFR 50.2: "Design bases mean 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 ...". The design bases of a facility, as so defined, is a subset of the licensing basis and is contained in the UFSAR. Information developed to implement the design bases is contained in other documents, some of which are docketed and some of which are retained by PECO Energy. The design basis for each facility forms the legal basis by which compliance with NRC requirements is to be judged.
Engineering Change Request (ECR)	Design change packages are developed and published as Engineering Change Requests (ECR) under the PIMS Document Control module. The text associated with the design change is contained in the computerized record while drawing and document revisions are hard copy attachments to the ECR. These hard copy attachments and the computer printouts are treated as nuclear records.
Licensing Bases	The licensing bases includes NRC regulations, orders, license conditions, exemptions, and technical specifications. It also includes the plant specific design basis information as documented in the UFSAR and the commitments remaining in effect that were made in docketed licensing correspondence, such as responses to NRC Bulletins, Generic Letters and Enforcement Actions, as well as commitments documented in Licensee Event Reports.

ATTACHMENT 4

Glossary of Terms

Performance Enhancement Program (PEP)	Procedure LR-C-10 is the PECO Nuclear procedure that establishes the Performance Enhancement Program (PEP) to improve performance through evaluation of conditions adverse to quality and other enhancement opportunities, including trend information. The procedure provides direction for the identification and evaluation of issues to ensure they are thoroughly reviewed, including causal factor and generic implication identification with subsequent implementation of corrective actions to prevent recurrence.
Plant Information Management System (PIMS)	In order to enhance our control over work processes, PECO Nuclear moved to a common integrated computer based process called PIMS (Plant Information Management System). PIMS terminals are located at all PECO Nuclear facilities, providing controlled access to centralized data for all organizations. The engineering design functions are performed primarily in the Document Control, Management Action, Resource Data, and Inventory Control PIMS modules.
Potentially Reportable Item	A condition, event or situation which is not expected and does not meet nuclear safety or environmental requirements or which requires that a notification be made or that a non-routine report be written including but not limited to: • Unplanned, unexpected or possibly unanalyzed events or conditions, • Degradation, damage, failure, malfunction or inoperability of plant equipment, • Deviation from or deficiencies involving design basis requirements, licensing documents, QA requirements, or administrative controls; or • Planned press releases related to above.
Reportable Item	Any Potentially Reportable Item that has been determined to be reportable (i.e., requiring a prompt or verbal notification or a non-routine report).
SAR	The documents submitted to, or received from the NRC, which contain the Safety Analysis for SSCs in support of maintaining a Nuclear Power Facility Operating License. It includes information that describes the facility and presents the design basis and the limits of its operation as well as the safety analysis of the facility as a whole.

ATTACHMENT 4

List of Acronyms and Abbreviations

A/E - Architect Engineer
ARI - Alternate Replacement Item
ASAR - Annual Summary Assessment Report
BOM - Bill of Material
BWROG- BWR Owners Group
CAQ - Condition Adverse to Quality
CEMS - Critical Equipment Monitoring Systems
CM - Configuration Management
CRL - Component Record List
CTP - Commitment Tracking Program
DA - Design Authority
DBD - Design Baseline Document
DCR - Design Change Request
DEC - Design Equivalent Change
DID - Design Input Document
EA - Engineering Assurance
ECCS - Emergency Core Cooling System
ECR - Engineering Change Request
ECW - Emergency Cooling Water
EDS - Electrical Distribution System
EDSFI - Electrical Distribution System Functional Inspection
ENS - Emergency Notification System
EOP - Emergency Operating Procedure
EP - Emergency Preparedness
EPG - Emergency Procedure Guideline
EQAB - Engineering Quality Achievement Board
ESP - Engineering Support Personnel
ESPCT - Engineering Support Personnel Continuing Training
ESW - Emergency Service Water
ETT - Equipment Trouble Tag
FAC - Flow Accelerated Corrosion
FM - Fuel Management
FOL - Facility Operating License
GET - General Employee Training
HELB - High Energy Line Break
HPCI - High Pressure Coolant Injection
HPSW - High Pressure Service Water
I&C - Instrumentation & Controls
IISCP - Improved Instrument Setpoint Control Program
IM - Information Management
INDMS - Integrated Nuclear Data Management System
IPC - Inventory Parts Catalogue
IPEEE - Individual Plant Examination of External Events
ISCR - Instrument Setpoint Changes Request
ISEG - Independent Safety Engineering Group
ISI - Inservice Inspection
ISR - Information Service Request
IST - Inservice Testing

ATTACHMENT 4

List of Acronyms and Abbreviations

ITS	-	Improved Technical Specifications
JUMA	-	Joint Utility Management Audit
LCA	-	Licensing Change Application
LCO	-	Limiting Condition of Operation
LCR	-	Licensing Change Request
LER	-	Licensee Event Report
LGS	-	Limerick Generating Station
LOCA	-	Loss of Coolant Accident
LR	-	Licensing and Regulatory
LRE	-	Lead Responsible Engineer
MCC	-	Motor Control Center
MOD	-	Modification
MOP	-	Master Oversight Plan
MOV	-	Motor Operated Valve
MSRV	-	Main Steam Relief Valve
NCR	-	Nonconformance Report
NIS	-	Nuclear Information System
NOV	-	Notice of Violation
NPRDS	-	Nuclear Plant Reliability Data System
NQA	-	Nuclear Quality Assurance
NRB	-	Nuclear Review Board
NRMS	-	Nuclear Records Management System
NSAC	-	Nuclear Safety Analysis Center
NSHC	-	No Significant Hazards Consideration
ODCM	-	Offsite Dose Calculation Manual
OE	-	Operating Experience
OEAP	-	Operating Experience Assessment Program
PAD	-	Performance Assessment Division
PBAPS	-	Peach Bottom Atomic Power Station
PEP	-	Performance Enhancement Program
PI	-	Performance Indicators
PIMS	-	Plant Information Management System
PLCO	-	Potential Limiting Condition for Operations
PLS	-	Personal Librarian Software
PMS	-	Plant Monitoring System
PORC	-	Plant Operations Review Committee
PRI	-	Potentially Reportable Item
PSA	-	Probabilistic Safety Assessment
QR		Quality Reviewer
RC	-	Reportability Coordinator
RCIC	-	Reactor Core Isolation Cooling
RE	-	Regulatory Engineer
REM	-	Radiological Environmental and Meteorological Monitoring
RHR	-	Residual Heat Removal
RHRSW	-	RHR Service Water
RRM	-	Reportability Reference Manual
RS	-	Responsible Superintendent
SALP	-	Systematic Assessment of Licensee Performance
SAR	-	Safety Analysis Report

ATTACHMENT 4

List of Acronyms and Abbreviations

SBGT	-	Standby Gas Treatment
SCN	-	Stock Code Number
SER	-	Safety Evaluation Report
SME	-	Subject Matter Expert
SO	-	System Operating
SQR	-	Station Qualified Reviewer
SQUG	-	Seismic Qualification Utility Group
SR	-	Surveillance Requirement
SRV	-	Safety Relief Valve
SSC	-	Structure, Systems, and Components
SSD	-	Safe Shutdown
SSFI	-	Safety System Functional Inspection
ST	-	Surveillance Test
TC	-	Temporary Change
TCF	-	Troubleshooting Control Form
TECH SPEC	-	Technical Specification
TPA	-	Temporary Plant Alteration
TRIP	-	Transient Response Implementation Plan
TS	-	Technical Specification
TSA	-	Tech Spec Action
UFSAR	-	Updated Final Safety Analysis Report
USQ	-	Unreviewed Safety Question

ATTACHMENT 4

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