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United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
10 CFR 50.54(f) Submittal Regarding
Adequacy and Availability of
Design Bases Information

Ladies and Gentlemen:

The Centerior Energy Corporation, on behalf of the Cleveland Electric Illuminating Company, received a letter from the Nuclear Regulatory Commission (NRC) regarding the adequacy and availability of design bases information for the Perry Nuclear Power Plant (PNPP). The letter requested the following information:

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

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

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

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

(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.

The request further asked whether the licensee has undertaken any design review or reconstitution programs, and if not, to explain the rationale for not implementing such a program. This request will be treated as request (f).

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The Perry Nuclear Power Plant responses to the six (6) requests are contained in Attachments 1 through 6, respectively.

The Cleveland Electric Illuminating Company believes that the processes and programs as described in the attachments provide reasonable assurance that: design bases information has been translated into plant procedures, programs are in place to control and evaluate changes, systems perform and are tested appropriately, and any deviations identified are reconciled.

The information provided in the attachments is intended to describe the current processes, programs, and procedures. It is not intended to preclude subsequent changes following normal practices or to require NRC notifications or approvals of such changes other than those currently required. This letter does not modify any prior NRC commitments, but does create the following new commitments:

1. The effectiveness of the overall Design and Configuration Control Processes at PNPP will continue to be evaluated through a pattern of Safety System Functional Assessments (SSFA). This type of assessment is planned to be implemented at a frequency of one (1) SSFA per reactor operating cycle on risk significant systems/trains/functions following the scoping criteria contained in the PNPP Maintenance Rule Program.

2. The existing PNPP Configuration Management Improvement Program (CMIP) initiatives have been focused primarily in the Perry Nuclear Engineering Department. PNPP will establish a site-wide administrative procedure and revise existing plant procedures/instructions to reflect the Configuration Management hierarchy. This commitment is intended to further assure that the processes and programs utilized to operate, maintain, and performance monitor SSCs important to safety accurately reflect the supporting design bases. PNPP plans to utilize some of the guidelines contained in NUREG-1397, "An Assessment of Design Control Practices and Design Reconstitution Programs in the Nuclear Power Industry", and NUMARC 90-12, "Design Basis Program Guidelines", to further enhance the CMIP. The plan for accomplishing this activity will be transmitted to the NRC under separate cover letter by August 31, 1997.

If you have questions or require additional information, please contact Mr. Henry L. Hegrat, Manager - Regulatory Affairs, at (216) 280-5606.

Very truly yours,

HRR/CRA/HBO/JEE:sc

Attachments

cc: NRC Region III
Director, Office of Nuclear Reactor Regulation
NRC Resident Inspector
NRC Project Manager

RESPONSE

TO

10 CFR 50.54(f) REQUEST FOR INFORMATION

FOR THE

PERRY NUCLEAR POWER PLANT

This letter is submitted pursuant to 10 CFR 50.54(f). Attached is The Cleveland Electric Illuminating's response to the "Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Adequacy and Availability of Design Bases Information" for the Perry Nuclear Power Plant.

By: ___________________________
John P. Stetz, Senior Vice President - Nuclear

Sworn to and subscribed before me, the 14th day of February, 1997.

____
Notary Public, State of Ohio

SHERYL A. ECKELBERRY
Notary Public, STATE OF OHIO
My commission expires 5-9-1999
Recorded in Lake County

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

The response to this request will provide a brief description of the Perry Nuclear Power Plant (PNPP) processes in the areas of design and configuration control, the Updated Final Safety Analysis Report (USAR), and the Quality Assurance Program. These processes were developed in accordance with the applicable regulations. The foundation of these processes have been in place for the life of PNPP, including the design and construction phase of the plant.

Procedures and Instructions Overview

An overview of the PNPP procedure and instruction development process will be provided to show the hierarchy and the controls associated with the development and maintenance of the administrative controls associated with the design and configuration control, USAR, and Quality Assurance processes.

The PNPP Operations Manual contains the policies, plans, procedures, and instructions which collectively provide the methods to perform and manage the activities of PNPP. The two basic document types comprising the PNPP Operations Manual are Plant Administrative Procedures (PAP) and instructions. The PAPs are upper-tier documents which implement the requirements of the Technical Specifications, the USAR, the Quality Assurance Program, national codes and standards, regulations, licensing commitments, or items which are considered important. As part of the development or revision of each PAP, a 10 CFR 50.59 review is required to be performed. The PAPs also require review by the on-site review committee, Plant Operations Review Committee (PORC), prior to approval. The scope of the PAPs may effect multiple departments and are of major significance to assure safe and efficient plant operation. PAPs establish responsibilities, lines of authority, and methods of controlling a specific activity or area of activity. Instructions are subordinate to the PAPs. Instructions are limited in scope and are used to implement the requirements of the upper-tiered documents. Instructions provide detailed steps necessary for the performance of an activity including steps required to operate, test, or maintain equipment or components; to provide for data collection and analysis activities; or to implement a process. As part of the development and revision of most instructions [e.g. System Operating Instructions (SOI), Surveillance Instructions (SVI), Nuclear Engineering Instructions (NEI)], a 10 CFR 50.59 review is required to be performed. Select instructions require PORC review prior to approval. The process for writing procedures and instructions consists of an individual knowledgeable in the subject matter who prepares the document; the performance of a series of interface or cross-functional reviews of the document, if appropriate; and an in-depth review of the document performed by an individual who is trained and certified to perform this review. These processes are documented in PAP 0501, "Perry Operations Manual", PAP 0502, "Preparation, Review, and Approval of Procedures", and PAP 0507, "Preparation, Review, and Approval of Instructions."

Design and Configuration Control

The PNPP employs a number of processes for developing and maintaining design bases requirements, design output documents, plant physical configuration

and facility documentation. Fundamental to PNPP's design and configuration control philosophy are the processes for the preparation, revision, review and approval of procedures and instructions used to operate, test, maintain and modify Systems, Structures and Components (SSC).

The processes which control engineering and design activities at PNPP are derived from, but not limited to, the requirements of 10 CFR 50, Appendix B, Criterion III; ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants"; ANSI N18.7, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"; and the PNPP Quality Assurance Plan. The process requirements are reflected in procedures developed to implement the requirements of the applicable regulations and standards. These engineering and design processes focus on the control of plant modifications and the maintenance of configuration documentation related to the day-to-day functioning of the plant.

Design Control

Permanent modifications to SSCs under the control of the configuration management process are prepared, reviewed and approved in accordance with processes outlined in PAP 0309, "Processing Plant Modifications." This procedure describes the processes for both major and minor modifications (called modifications and simple modifications, respectively). Nuclear Engineering Instructions (NEI), which are subject to the process controls for instruction development and revision as described above, provide the detailed instructions for implementing the higher tiered PAP 0309 requirements.

The distinction between major and minor modifications is based on the magnitude (size/cost), complexity (e.g., number of system interactions), and degree of design difficulty (e.g. amount of analysis, number of interdiscipline interfaces) of the proposed modification. The processes for both major and minor modifications are governed by a detailed instruction (NEI 0330, "Interface Reviews and Evaluations") to determine and conduct interdiscipline design interfaces. NEI 0330 utilizes an Interface Review Checklist (IRC) as a method of establishing the design bases impact of a proposed change. An affirmative response to any question posed on the checklist initiates an interface evaluation by the engineering discipline assigned the responsibility for maintaining and interpreting that specific portion of the design bases. The interface evaluation identifies/describes the original design bases for the SSC, the effect of the proposed modification on the original design bases, relevant design criteria and standards, potential failure mechanisms and failure consequences, and other pertinent considerations associated with the proposed modification. Required interfaces and evaluations are summarized and retained with each design change. For major modifications, the process also requires the establishment of a detailed design report. The design report is the focal point for establishing the design bases of the SSC prior to modification and for identifying and reconciling the effect the change has on the established design bases. Guidance provided for completion of the design report specifically refers the preparer to the requirements of ANSI N45.2.11 for establishing the parameters, bases, and inputs upon which the design is based. The design change package receives an interface review from those work groups which had input to its development. This review focuses the work groups on the completed design change package at the stage in which the interfaces are assembled in an

integrated fashion. Modifications affecting an SSC designated as safety-related or augmented quality receive a complete ANSI N45.2.11 independent design verification following the integrated review.

In order to assure that the appropriate operations, maintenance, and testing documents are revised as a result of implementing a design modification, NEI 0373, "Initiating, Developing, and Processing Design Modifications", and NEI 0374, "Simple Modification Requests", require the development of a Modification Documents List (MDL). The MDL is part of the modification package and identifies the affected drawings and documents that require revision or initial issue as a result of implementing the modification. The list includes the applicable operations, maintenance, and testing procedures and instructions. As part of the modification closure process, personnel are required to assure that documents associated with the modification have been appropriately processed. As a second check to this process, operations and instrumentation and controls personnel are required to sign-off the modification package indicating that the appropriate procedures and instructions have been revised or issued in concert with the implementation of the modification.

The control processes for permanent modifications require the performance of a 10 CFR 50.59 review as part of the development of the modification.

Configuration Control

The PNPP configuration control processes are similar to the model depicted in NUMARC 90-12, "Design Basis Program Guidelines", October 1990, focusing on the design bases of the facility and communication of these requirements to the processes which operate, test, and maintain the plant. The PNPP Configuration Management Improvement Program (CMIP) identifies the SSCs that are subject to PNPP configuration control policies and procedures. The CMIP identifies the PNPP document types, data bases and computer applications that contain configuration information. Additionally, the CMIP identifies the primary configuration control processes implemented at PNPP as well as the assessment processes that are in place to monitor implementation of configuration management.

In addition to managing configuration control associated with design changes, the CMIP identifies control processes for the maintenance of configuration documentation in the day-to-day functioning of the plant. Examples of these processes are:

- setpoint changes,
- computer software administrative control, and
- drawing change control.

The program for design control of setpoints is contained in PAP 1403, "Control of Setpoints." The program utilizes the Setpoint Change Request (SCR) as the mechanism for implementing setpoint changes. The SCR process encompasses the preparation, review, and approval of the setpoints. The SCR process requires identification of design inputs, performance of design verification, and performance of 10 CFR 50.59 reviews, if appropriate. The procedure requires additional reviews to be performed on work documents and other plant procedures/instructions to assure the setpoints are accurately reflected in the

plant. PAP 1403 also provides the mechanism for updating the Master Setpoint List which maintains configuration control of both existing and new setpoints.

The process controls for computer software are delineated in PAP 0506, "Computer Software Administrative Control." This procedure applies to software which affects the operation, design, or safety of the plant. The control process for safety-related software development/modification provides for a description of the design requirements, a verification and validation plan performed by an independent party, and a 10 CFR 50.59 review. Software testing, installation, and back-up activities are also included in PAP 0506.

A Drawing Change Notice (DCN) is a design change document prepared to revise or modify a previously approved design drawing. The DCN process is described in NEI 0363, "Drawing Change Notice." DCNs can be classified as either editorial or design/arrangement. An editorial DCN is used to show a change, addition or deletion which does not affect the design configuration as depicted on the drawing. A design/arrangement DCN is used to show a change to the design configuration of an SSC or to provide alternate equivalent configurations. Process controls for both editorial and design/arrangement DCNs require the performance of a 10 CFR 50.59 review. A design/arrangement DCN requires a multi-disciplinary review and a verification, if the drawing is safety-related or an augmented quality critical drawing. The identification of the reviewing disciplines is performed through the use of the Design Interface Summary (DIS) sheet.

These processes, as with the other processes used for day-to-day design bases maintenance, require the performance of 10 CFR 50.59 reviews (other than for editorial or administrative issues), as well as design review/verification depending on the safety classification of the change.

Oversight of the design and configuration control processes is governed by the Quality Assurance Program through the audit and surveillance functions performed by the Perry Quality and Personnel Development Department. Additional oversight of the various engineering activities performed at PNPP is conducted by the Independent Safety Engineering Group (ISEG) and by the Engineering Assessment Review Team (EART).

If potential issues are found, the issues are documented, evaluated and tracked by the PNPP Corrective Action Program.

<u>Updated Final Safety Analysis Report (USAR)</u>

The USAR for the PNPP was submitted as an update of the PNPP Final Safety Analysis Report (FSAR) in March 1988. The USAR was prepared and submitted in accordance with 10 CFR 50.71(e). The USAR provides a description of PNPP Unit 1, common facilities, and those Unit 2 facilities needed to support Unit 1 operations. The content and format of the Perry USAR follows that of the FSAR and the guidelines contained in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition", Revision 3. In October 1996, PNPP submitted Revision 8 of the USAR.

The administrative controls for the USAR are contained in two plant documents: PAP 0520, "Changes to Licensing Documents and License Amendments", and Licensing

and Compliance Instruction (LCI) 0605, "PNPP Change Request and USAR Revision Processing." USAR Change Requests are supported by a 10 CFR 50.59 Safety Evaluation. The exception to this is for those activities that have been previously reviewed and approved by the NRC. Change requests reflecting these NRC approved activities are supported by the NRC approval. Change requests are subjected to a site peer review process to assure the various work groups impacted by the proposed USAR changes are cognizant of the changes and have input into the change request prior to its approval. Additionally, change request packages are reviewed against the PNPP Technical Specifications and licensing commitments contained in the Perry Regulatory Information Management System. For those changes in USAR Chapter 17, "Quality Assurance", the change request receives a review in accordance with 10 CFR 50.54(a). If the USAR change is deemed to be a reduction in commitment of the Quality Assurance Program, the USAR change is held until a request is approved by the NRC to allow the change. When the change request is approved, it is then considered part of the PNPP USAR.

NEI 0373, "Initiating, Developing, and Processing Design Modifications", and NEI 0374, "Simple Modification Requests", require a draft USAR Change Request as part of the design package if the design change impacts the contents of the USAR. Additionally, PAP 0305, "Safety Evaluations", provides guidance to the preparers of 10 CFR 50.59 Safety Evaluations to revise the USAR, if necessary, based upon the activity the 10 CFR 50.59 Safety Evaluation was written against.

USAR reviews are required in the performance of various site work activities (e.g. design modifications, procedure changes, and preparation of 10 CFR 50.59 evaluations). To aid in performance of this USAR review, a computerized USAR database was developed and came on-line in 1993. This database allows individuals to perform word searches on the USAR to determine potential USAR impacts upon the activity being evaluated.

Oversight of the USAR process is governed by the Quality Assurance Program through the audit and surveillance function performed by the Perry Quality and Personnel Development Department.

If potential discrepancies in the USAR are found, the discrepancies are documented, evaluated and tracked by the PNPP Corrective Action Program.

10 CFR 50.59 Program

The 10 CFR 50.59 Program at the PNPP is a two-step process. The first step is a screening evaluation to determine if a change to the plant, change to a procedure, or performance of a test not described in the USAR exists. The second step provides for the evaluation to determine if an Unreviewed Safety Question (USQ) exists. The program is described in PAP 0305, "Safety Evaluations."

The screening document is called an Applicability Check. The Applicability Check asks five questions. Questions 1 through 3 are identical to the statements described in 10 CFR 50.59(a)(1). Question 4 is used to determine if the activity under evaluation changes the Technical Specifications. Question 5 is used to determine if the activity under evaluation impacts the Environmental Protection Plan (EPP) contained in the PNPP operating license. If Questions 1, 2, or 3 are answered "Yes", then a Safety Evaluation is required to be prepared.

If Question 4 is answered "Yes" and management decides to pursue the activity, then a license amendment will be prepared pursuant to 10 CFR 50.90. If Question 5 is answered "Yes", then an Environmental Evaluation is prepared pursuant to the EPP.

The Safety Evaluation determines if an USQ exists. The Safety Evaluation asks seven questions. The questions are derived from the three questions detailed in 10 CFR 50.59(a)(2).

Two notable features of the PNPP 50.59 Program are the definitions associated with the terms "Safety Analysis Report" and "Accident." "Safety Analysis Report" is defined as the USAR (along with changes that have been approved but not yet incorporated into the hardcopy USAR), the Quality Assurance Program, the Emergency Plan, the Fire Protection Program, the EPP, documents referenced within the body of the USAR, and the Nuclear Regulatory Commission's Safety Evaluation Report. "Accident" is defined as the analyzed events with which the plant was designed to cope, and includes anticipated operational occurrences, abnormal operational occurrences, accidents, other hypothetical events, and external events. The events in question are contained in various USAR sections.

Applicability Checks and Safety Evaluations may only be prepared by personnel who are trained to perform this function. The training program includes classroom training with a written examination and a required reading list. Part of the classroom training includes an USAR overview and an accident analysis overview. Each Applicability Check and Safety Evaluation is independently reviewed by an individual who has the same training as the Applicability Check/Safety Evaluation preparer.

The PORC reviews each Safety Evaluation and renders a recommendation to senior plant management for approval/disapproval.

The following are examples of activities that require the performance of an Applicability Check:

- Procedures and Instructions contained in the PNPP Operations Manual,
- Potential Issue Forms describing nonconforming conditions with a "use-as-is" or "repair" disposition that do not change design documents,
- Drawing Change Notices,
- Design Change Packages, and
- Specification Change Notices.

Oversight of the 10 CFR 50.59 Program is provided through the Quality Assurance Program through audits and surveillances. Additional oversight is provided by the off-site review organization, the Company Nuclear Review Board (CNRB). The CNRB requires that each Safety Evaluation be reviewed to determine if an USQ exists.

If potential discrepancies in the 50.59 program are noted, the discrepancies are documented, evaluated and tracked by the PNPP Corrective Action Program.

Quality Assurance Program

USAR Section 17.2, "Quality Assurance During the Operations Phase", provides the docketed Nuclear Quality Assurance Program for the PNPP. The PNPP Nuclear Quality Assurance Program complies with the requirements of 10 CFR 50, Appendix B. Each of the eighteen criteria of 10 CFR 50, Appendix B, and the responsibilities for implementing the corresponding activities, are addressed in detail in the eighteen subsections which comprise USAR Section 17.2. The responsibility for the implementation of the Nuclear Quality Assurance Program is interdepartmental and not the sole responsibility of any organization or group.

The Nuclear Quality Assurance Program is implemented through the PNPP Quality Assurance Plan. The Quality Assurance Plan defines specific individual and organizational responsibilities and authority for the Nuclear Quality Assurance Program. It also prescribes the requirements for compliance with the appropriate regulations.

The Quality Assurance (QA) Plan addresses the 18 criteria contained in 10 CFR 50, Appendix B. Section 18 of the QA Plan, "Audits and Surveillances", describes the actions of the Quality Assurance Section to assess and verify compliance with the PNPP Nuclear Quality Assurance Program. The QA Plan states that a comprehensive system of planned and periodic audits and surveillances shall be established to assure the following:

- Quality assurance requirements are adequate, effective and implemented.
- Conditions adverse to quality are identified and corrected.
- The PNPP Nuclear Quality Assurance Program complies with USAR commitments to Regulatory Guides, ANSI standards, and other codes and standards.
- Data is provided for continuing evaluation of the effectiveness of the PNPP Nuclear Quality Assurance Program.

These QA audits and surveillances also verifies compliance and effectiveness of the following:

- PNPP Quality Assurance Plan.
- Regulations, license provisions, and Plant Technical Specifications.
- Procedures and instructions which describe or control activities within the scope of the QA program.
- Programs for training, retraining, qualification and performance of operating personnel.
- Corrective actions program.
- Observation of performance of operating, refueling, maintenance and modification activities, including associated record keeping.

The QA Plan provides audit/surveillance subject areas, frequencies, procedures for planning, performing, and reporting audits, and corrective action requirements. QA audits/surveillances are administered through the Nuclear Quality Instructions (NQIs). NQI 1801, "Audit Program Control", describes the requirements for performance of systematic program audits to assess compliance with and effectiveness of the PNPP Nuclear Quality Assurance Program. NQI 1801 describes the methods of planning, conducting, and reporting quality assurance program audits. NQI 1802, "Surveillance Program Control", establishes the administrative controls for the operational surveillance program. NQI 1802 describes the planning, performance, and reporting of operational surveillances.

Audit/surveillance frequencies and more detailed descriptions are located in a document titled the Functional Area Planning Guide (FAPG). The FAPG incorporates audit and surveillance requirements contained in various documents (e.g. the USAR, Technical Specifications, Code of Federal Regulations).

Section 7.0 of the FAPG, "Engineering and Technical Support", states that an audit of design control (10 CFR 50, Appendix B, Criterion III, "Design Control") shall be performed every two years. The scope of the Design Control audit includes design activities, including design input and maintenance of design basis, design verification, organizational interfaces, initiation/processing of design changes, and control of design documents. The audit also evaluates preparation and implementation of design modifications including closure activities. USAR compliance/updates and 10 CFR 50.59 Applicability Checks/Safety Evaluations are covered by this audit. The FAPG lists some of the major activities of the Design Control audit:

- Design inputs such as design bases, regulatory requirements, codes, standards, and commitments are identified, documented and approved. Actual use of design inputs is also verified.
- Completeness and correctness of design analysis assumptions, calculations, use of reference material, proper design objectives, checking computer calculations, documentation, and proper review and approvals (design interface summary and evaluations).
- Correct use and control of design documents, such as drawings and specifications.
- Design verification is independent and effective to substantiate the adequacy of a design, calculation or analysis.
- Design output documents are correctly prepared, approved, and issued.
- 10 CFR 50.59 Applicability Checks and Safety Evaluations are adequately performed.
- Verify design modifications are properly approved and address original design intent.
- Verify design modifications are properly implemented, closed, and reflected through the use of the appropriate configuration controls.
- Verify that the USAR and Technical Specifications are updated.
- Verify the design process establishes controls to assure that design features specified in the USAR are not changed without prior NRC approval and proper amendments to the Operating License, Technical Specifications, and the USAR.

In addition to the Design Control audit, surveillances are performed on:

- Engineering Interface: Verify that field clarification requests, nonconformance Potential Issue Forms, and plant operability evaluations are properly processed. The evaluation should include response timeliness, impact on design basis/documents, steps to prevent recurrence, and document revision.
- Design Changes: Verify that changes to Design Change Package designs are properly processed.
- Setpoint Change Request: Verify that setpoint change requests are properly processed and reviewed for impact on system parameters.

- Design Assurance: Design modifications should be reviewed for proper interface reviews, verification, and evaluations when required. The modifications should be reviewed for conformance to codes and standards, agreement with original design basis, and impact on plant procedures.
- Design Change Package Closeout: Modification packages should be reviewed to determine that drawing updates, and procedure/instruction changes have been made.

Section 8.0 of the FAPG, "Safety Assessment/Quality Verification", states that an audit of the effectiveness of corrective action shall be performed every six months. The scope of the Effectiveness of Corrective Action audit includes the completion, adequacy, timeliness and effectiveness of management controls for implementing corrective actions. The audit evaluates deficiencies occurring in unit equipment, structures, systems, and methods of operation affecting nuclear safety, Potential Issue Forms, Licensee Event Reports, Stop Work actions, and trend analysis. The FAPG lists the major activities of the Effectiveness of Corrective Action audit:

- Problems are promptly identified.
- Significant problems are reported and evaluated.
- Deficiencies are evaluated and dispositioned.
- Immediate or compensatory actions are taken, as appropriate.
- Full technical justification is given for use-as-is dispositions for nonconformances.
- Ensure resolutions include actions to prevent recurrence.
- Verify actions to prevent recurrence are effective.
- Verify adverse trends are identified through trending and evaluated for any follow-up actions.
- Verify adverse trends effecting safety are acted upon in a timely manner.
- Determine cause and document problems for conditions where actions are necessary/desirable to prevent recurrence.
- Determine ability to identify and document possible causes of problems.
- Take effective corrective action on the most probable cause determined.

The Nuclear Quality Assurance Program description and the Quality Assurance Plan are maintained and revised in accordance with NQI 0203, "Maintenance of the Quality Assurance Program." Changes to the QA Plan are reviewed for compliance with the USAR and PNPP commitments. If the change to the QA Plan requires a change to the USAR, a USAR change request is submitted in accordance with PAP 0520. Changes to the Nuclear Quality Assurance Program descriptions contained in USAR Section 17.2 are reviewed to assure compliance with the criteria of 10 CFR 50 Appendix B, and to identify any reduction in Nuclear Quality Assurance Program commitments. If the change involves a reduction in a Nuclear Quality Assurance Program commitment, a letter requesting NRC approval in accordance with 10 CFR 50.54(a) is submitted. Processing of Quality Assurance Plan changes which involve USAR changes are deferred until the USAR Change Requests are approved by the NRC. Quality Assurance Plan changes require the approval of the Manager, Quality Assurance Section and the Vice President - Nuclear, Perry.

Oversight of the Quality Assurance Program is governed through the Nuclear Quality Assurance Program's audit and surveillance function. Additional oversight is provided by the CNRB and the Joint Utility Management Assessment

(JUMA) Program (JUMA was created to provide independent peer reviews of participating utility quality assurance programs). JUMA has been used to perform audits of the Nuclear Quality Assurance audit and surveillance program.

If potential discrepancies in the Quality Assurance Program are found, the discrepancies are documented, evaluated and tracked by the PNPP Corrective Action Program.

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

The response to this request will briefly describe the programs and processes which were used to translate the design bases into operating, maintenance, and testing requirements. An overview of the various inspections and assessments that have occurred at the Perry Nuclear Power Plant (PNPP) since the start of commercial operation in 1986 will provide the rationale that the design bases have been translated into the operating, maintenance, and testing requirements. From these inspections and assessments, it can be concluded that the design bases have been adequately translated into the relevant operating, maintenance, and testing documents and processes. This is supported by the small number of identified problems which potentially could have impacted System, Structure, and Component (SSC) functionality. However, weaknesses have been identified in the area of design basis documentation thoroughness (e.g. calculation detail, rigor of design reviews). These weaknesses provide the impetus for the engineering improvement initiatives currently in progress. These initiatives are described in Attachments 5 and 6.

Programs/Processes

In August 1985 (prior to the issuance of PNPP's operating license) Special Project Plan 0501 (SPP 0501), "Operations Manual Verification", was initiated to provide additional assurance that plant operational commitments were identified and incorporated into the PNPP Operations Manual. This plan included a review of certain Operations Manual instructions (i.e. Integrated Operating, System Operating, Alarm Response, Off Normal, Plant Emergency, Valve Line Up, and Electrical Line Up Instructions) to assure that they were consistent with the as-built configuration of the plant.

The first step in the commitment verification portion of this plan was the performance of a search of the Final Safety Analysis Report, the Technical Specifications, the NRC Safety Evaluation Report, Regulatory Guides, etc. to identify commitments which involved administrative controls, plant operations and maintenance, testing, inspections, radiological controls, and personnel qualifications and staffing. An evaluation was then performed to determine if the identified commitments were adequately addressed in the Operations Manual. The Perry Regulatory Information Management System (the commitment tracking system) was used to assure that commitments were tracked to closure. All commitments requiring closure for fuel load were closed. Additionally, any system related commitments that were identified as part of the commitment review were closed with the System Operability Verification Checklist of Special Project Plan 1028 (SPP 1028), "Fuel Load Achievement."

The review of instructions utilized a baseline of system turnover to the site's technical and operating organizations, at which time the instructions were reviewed against the as-built drawings for each system. Any changes needed to the aforementioned instructions, as a result of this review, were incorporated in accordance with the applicable plant procedures. To assure timely completion of this process, the instruction review was also tracked by SPP 1028. Before any system was declared operational, the necessary instructions were required to be verified and revised. This review was completed in June 1986.

These review programs provided a measure of assurance that the plant's operating, maintenance, and testing instructions were consistent with the as-built plant configuration.

Changes made to these documents after this "baseline" program were made in accordance with the applicable Operations Manual procedures. Attachment 1 provides an overview of this process. Due to the controls associated with the procedures/instruction revision processes (e.g. 10 CFR 50.59 reviews, cross-functional reviews, Plant Operations Review Committee (PORC) review) and the design control processes (refer to Attachment 1), a further measure of assurance exists that the applicable operating documents remain consistent with the plant configuration.

Internal Inspections/Assessments

The inspections and assessments summarized below were selected because they were vertical slice reviews of design bases documentation. Included as part of the process was a review of design bases documentation to verify appropriate translation into plant operational documents.

1. A Safety System Functional Inspection (SSFI) was conducted on the Emergency Service Water System in February 1989. The methodology utilized for this SSFI included, but was not limited to, an examination of design bases documents to determine limiting and operational system parameters, and a comparison of procedures and test results to the design bases. The issues identified in the SSFI were determined not to impact operating, testing, or maintenance procedures/instructions as related to design bases information. Therefore, the Emergency Service Water System would continue to perform its intended function.

2. A SSFI was performed on the Suppression Pool Make-up System in September 1990. This inspection employed a "vertical slice" approach to examine the effectiveness of eight (8) functional areas: design bases, design control, operations, maintenance, surveillance, testing, training, nuclear experience, and quality assurance.

   The design bases/design control area of the inspection performed a sampling review of calculations, Drawing Change Notices (DCN), Design Change Packages (DCP), and nonconformance reports associated with the Suppression Pool Make-up System. Several design documentation issues were identified; however, the significance of the issues did not indicate the Suppression Pool Make-up System could not perform its intended design function.

   The operations area of this inspection consisted of a review of various operational instructions, including System Operating Instructions, Alarm Response Instructions, Valve Lineup Instructions, and Temporary Test Instructions. The review also evaluated information contained in the Plant Data Book relative to the Suppression Pool Make-up System. From the operations standpoint, several minor concerns were identified. However, these concerns were evaluated as not related to the translation of design basis information to operating documents.

The maintenance/surveillance area of this inspection utilized a sampling approach to review the substantial number of maintenance work orders associated with the Suppression Pool Make-up System. The repetitive tasks, applicable Reliability Information Tracking System data, Nuclear Plant Reliability Data System data, and surveillance tests associated with the Suppression Pool Make-up System were also reviewed. For the most part, concerns identified within this area were related to the administration of activities and not to design bases information.

None of the concerns identified in this SSFI, either generically or individually, were such that the system would not perform its intended design function.

3. In October 1992, four members of the BWR6 Owners Group conducted a Safety System Functional Assessment (SSFA) of the High Pressure Core Spray (HPCS) and the Division III Diesel Generator (DG) Systems. SSFAs were performed at PNPP and at three other stations to assess the operational readiness of the above systems. The areas reviewed were mechanical, electrical, instrumentation and controls, operations, testing, and maintenance (the electrical design area was not reviewed at PNPP due to the Electrical Distribution System Functional Inspection (EDSFI) previously performed by the NRC).

   The primary focus of the PNPP report was to provide a compilation of the various issues and concerns from the HPCS and DG SSFAs of the four (4) participating BWR6s and evaluate them for their relevance to PNPP through an assessment observation and a design basis review. Examples of the types of issues and concerns evaluated were design calculation support, preventative maintenance, testing measurement inaccuracies, and valve-lineup clarifications. Based upon the areas evaluated, the PNPP report concluded the operational readiness of the HPCS and the Division III DG Systems was assured.

4. A more recent SSFA was performed by an off-site contractor in May 1993 on the Low Pressure Core Spray (LPCS) and the Reactor Core Isolation Cooling (RCIC) Systems. This SSFA utilized an assessment plan which provided the framework to answer the following:

   - How is the system operated compared with how it was designed to operate?
   - Have modifications since the plant was licensed altered the design in a manner such that it may not function as expected?
   - Are system components properly maintained?
   - Does post-modification and post-maintenance testing confirm the readiness of the system if called upon?
   - Does surveillance testing confirm the readiness of the system if called upon? Do acceptance criteria accurately reflect the design bases?
   - Have the operators been properly trained to operate the system? Are modifications accurately reflected in training documents?
   - Are management control programs effective to ensure that the system will function on demand?

This assessment reviewed a variety of documents which included but were not limited to the Technical Specifications, surveillance instructions, system ope-ating instructions, emergency procedures, and maintenance documents.

The assessment rated the surveillance program as good. A strength was noted in that a clear trail existed between engineering information and the surveillance test acceptance criteria. The assessment found the LPCS and RCIC Systems were being maintained in an adequate condition such that the two systems could perform their intended functions. The post-maintenance testing was determined to be adequate for verifying that the maintenance performed did not affect component function or operability. Additionally, the assessment did not note any concerns with respect to the operations area.

The assessment did note some weaknesses (e.g. ineffective root cause analyses, calculations not consistently being updated/superseded) in the design area. However, based upon the results of this SSFA, the LPCS and RCIC Systems were determined to be functional and could perform their intended safety function.

5. In October 1993, the Perry Course of Action (PCA) was implemented as a strategic plan to upgrade overall plant performance. The PCA identified the requirement to perform a System Operation and Test Review (SOTR) Program. The purpose of this program was to conduct a direct, deliberate confirmation of system and equipment readiness and functionality for three (3) systems that were representative of those forming an integral part of PNPP's operational capabilities. The systems represented an emergency system (High Pressure Core Spray), a reactor control and power generation system (Turbine Control and Steam Bypass), and a process system for radioactive waste (Off Gas). The SOTR review scope included:

   - Development and verification of detailed functional requirements,
   - Review of maintenance, modification history, and acceptance testing records,
   - Review of surveillance and periodic testing,
   - Identification of additional testing, and
   - Functional testing of the selected systems.

   The SOTR Program Final Report concluded that there was reasonable assurance that the systems evaluated were capable of performing as required, to support overall plant operation. Weaknesses : entified by the SOTR were not considered critical to system operability

6. A Systems Based Instrumentation and Control Inspection (SBICI) was conducted in accordance with NRC Inspection Procedure 93807, "Systems Based Instrumentation and Control Inspection", in March 1995. This inspection included a detailed review of the design and field installation of the associated instrument and control systems, setpoint calculations, mechanical system interfaces, calibration procedures, testability, isolation and bypass status indicators, maintenance, and equipment

installation. The instrument loops that were selected for this inspection are listed below:

- Reactor Pressure Vessel (RPV) Level, High Pressure Core Spray (HPCS) Initiation;
- Drywell High Pressure, HPCS Initiation;
- Degraded Voltage Protection;
- Condensate Storage Tank (CST) Level, Transfer to Suppression Pool;
- Low Pressure Core Spray (LPCS) Minimum flow;
- Main Steam Line (MSL) Tunnel High Ambient Temperature;
- Diesel Generator (DG) Starting Air Pressure;
- Main Steam Line High Flow;
- Suppression Pool Temperature; and
- Scram Discharge Volume Level.

The inspection concluded that the systems and instrumentation reviewed had their design bases well defined, setpoints appropriately calculated, supporting procedures and instructions in place and correct, and would perform their intended function. Concerns were identified with setpoints and setpoint methodology utilized for grid degraded voltage protection being nonconservative, CST freeze protection for level sensing impulse lines, and design bases behind the DG air start permissive setpoints. These concerns were satisfactorily resolved with design bases documentation updates, a minor setpoint change (grid degraded voltage), and a minor modification to further reduce the risk of CST instrument line freezing. An observation was made that, for some design areas, there was a lack of understanding by engineering personnel of the pertinent design bases. Subsequent training in 1996 focused, in part, on the area of design bases.

7. The Independent Safety Engineering Group (ISEG) initiated a Setpoint Applications Assessment (ISEG Project 96-110) in September 1996. The scope of this project was limited to those activities which supported design changes and implementing Master Setpoint List settings. For several safety-related setpoints, the entire setpoint process was reviewed (i.e. the process was evaluated from the design bases documents to final implementation in the field). Vertical slice assessment techniques were used during the performance of this assessment. The project concluded the PNPP setpoint application process was adequate and that there was reasonable assurance the evaluated components would perform their safety function.

   Although no SSC operability issues were found, some setpoint design process controls were found to be weak. Some setpoint related documentation was also found to be lacking detail, inaccurate, or hard to follow. These concerns are currently being tracked and evaluated within the PNPP Corrective Action Program.

8. In November-December 1996, ISEG performed an Emergency Service Water System Operational Performance Inspection (ESWOPI). This inspection included a review of the Emergency Service Water (ESW) System from the design, maintenance, operations, and training perspectives. The inspection involved the ESW design basis, system configuration, system operation with respect to design, maintenance, and testing to assure

component/system performance, single active and common mode failures, in-service testing, flooding, trending, fouling, and training. The inspection resulted in multiple findings, some of which may be generic in nature. Some of the findings include: a lack of adequate testing to demonstrate SSC functionality and calculation discrepancies/uncertainties. Evaluation of the findings is currently in progress as part of the PNPP Corrective Action Program.

External Inspections/Assessments

The adequacy of the PNPP programs and processes was supported by the NRC by their Diagnostic Evaluation Team (DET) evaluation in February-March 1989. The evaluation, in part, found that, before beginning operations, the plant met essentially all NRC licensing commitments and management had put in place technical programs, documents and processes which were considered fully supportive of operations.

An NRC Electrical Distribution System Functional Inspection (EDSFI) was conducted at PNPP during April-May 1991. The team considered the scope and implementation of the site program for surveillance testing of the Electrical Distribution System (EDS) a strength. Control of modifications for the EDS was acceptable and there appeared to be an adequate interface between engineering, operations, and maintenance. The team found the EDS and related support equipment to be properly installed and considered the material condition of the EDS a strength.

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

The response to this request provides the rationale for concluding that the configuration and performance of Systems, Structures, and Components (SSC) of the Perry Nuclear Power Plant (PNPP) remain consistent with the design bases.

The PNPP control processes for design and configuration control were briefly described in Attachment 1. The following will provide additional detail with emphasis on showing that SSC configuration and performance remains consistent with the design bases. Assessments of the PNPP processes as well as performance experience have indicated, in general, that SSCs have, and continue to be operated within the design bases of the plant.

<u>Configuration</u>

Systems are operated in accordance with specific operating instructions [e.g. System Operating Instructions (SOI), Integrated Operating Instructions (IOI), and Off Normal Instructions (ONI)]. These instructions are developed in accordance with the administrative controls described in PAP 0507, "Preparation, Review and Approval of Instructions." This procedure includes a process step to perform an "in-depth" review. This review is performed by personnel who are trained and certified to perform the function. The "in-depth" review requires the completion of a checklist that considers such items as: the incorporation of accurate limits/tolerances/acceptance criteria, equipment operation within its design envelope, and assurance of applicable industry code (e.g. American Society of Mechanical Engineers) conformance. The instruction preparer or "in-depth" reviewer also has the option to have any engineering discipline perform an interface review if deemed appropriate.

Prior to commencing refueling outages, a review of the outage schedule is performed to determine risk areas associated with the outage schedule. PAP 0115, "Outage Scheduling", and PAP 0116, "Shutdown Safety", provide the requirements to maintain a defense-in-depth by assuring more equipment is available than required by the Technical Specifications. Outage risk assessment software is used to assist in outage safety management. An important goal of the shutdown safety program is to maintain configuration control of the plant by assuring that systems are operated in accordance with approved instructions.

The following are additional activities that were performed at PNPP to assure that SSC configuration is known and consistent with the design bases:

1. In the mid 1980's, there were extensive walkdowns to address NRC Inspection and Enforcement Bulletin 79-14, "Seismic Analysis for As-Built Safety Related Piping Systems." Extensive walkdowns were also conducted, as part of a seismic clearance and anti-falldown program, to assure that seismic interactions of the as-installed SSCs did not impair the functioning of equipment important to plant safety or cause incapacitating injury to Control Room personnel. This program, developed from Regulatory Guide 1.29, "Seismic Design Classification", continues as an on-going program at PNPP. These walkdowns continue to assure that piping/piping supports were installed in accordance with design bases requirements and that unacceptable seismic interactions do not exist.

2. In 1984 and 1985, walkdowns of 10 CFR 50.49 equipment were performed by Quality Control personnel under the guidance developed by the Equipment Qualification (EQ) personnel, to verify that the equipment was located and installed in accordance with its applicable qualification documentation. The documentation from the walkdowns was reviewed and accepted by EQ personnel. Additionally, Quality Control personnel perform walkdowns/inspections of selected equipment during refueling outages to verify that conformance still remains with EQ requirements. If any discrepancies are identified, a corrective action document is initiated and dispositioned in accordance with the Corrective Action Program.

The Operations staff routinely monitors the configuration of plant components to assure they are properly aligned. Examples of these activities are:

1. The Operations staff routinely monitors the configuration of systems and components in the plant to assure proper alignment/operation. Plant equipment is monitored through the performance of PRI-TSR, "Technical Specification Rounds", and PRI-PER, "Plant Equipment Rounds." The PRI-TSR is used to document the performance of Technical Specification surveillance requirements. The PRI-PER is used to document the performance of plant equipment monitoring. In addition to obtaining specific data as delineated in the PRI-PER, the PRI-PER requires personnel to assess general area and equipment conditions. Any abnormal condition is reported, necessary corrective actions taken, and noted on the PRI-PER document.

2. The Process Computer System (PCS) provides monitoring of various inputs that represent significant plant process variables. The PCS scans these inputs at specified intervals and issues appropriate alarms and messages if limit or trip signals are received.

3. The Emergency Response Information System (ERIS) is an integrated system that gathers required data from systems/components throughout the plant. ERIS stores, validates, and processes that data. The system also generates visual displays for the operator on plant status information, provides printed records of transient events, and allows access to stored data for future analysis.

4. Plant annunciator alarms provide a visual and audible indication in the Control Room. These alarms indicate that key system parameters or configurations are not as expected. Alarms are promptly reported, investigated, and resolved in accordance with specific instructions for each annunciator window.

Design change implementation, interim system configuration control, and design change package closure requirements are contained in PAP 0309, "Processing Plant Modifications." This includes requirements to assure that the as-installed design change conforms to design output documents and design bases. Prior to system return to service/operable status, post-modification testing is typically required to demonstrate SSC functionality per the design intent of the modification. Such testing may be performed as part of a temporary test approved for that specific purpose, performance of an existing plant procedure/instruction, or through other controlled means.

One of the methods that can change the configuration of the plant outside of PAP 0309, "Processing Plant Modifications", is a temporary modification. Temporary modifications are short-term alterations required to support plant operation normally allowed to be installed for no longer than one (1) operating cycle, subject to further review. Control of temporary modifications is governed by PAP 1402, "Temporary Modification Control." This procedure establishes a method to assure operator awareness, conformance with the design intent/operability requirements, and preservation of plant safety and reliability. It provides the requirements for controlling and documenting short-term physical or functional changes to plant SSCs.

A technical evaluation is required to be performed on the proposed temporary modification to determine the effect upon such items as:

- operating procedures and instructions,
- design parameters, allowances, and interfaces,
- plant operating conditions, and
- plant limitations.

The technical evaluation is reviewed by design personnel to assure that the modification is consistent with design allowances and limitations. The Interface Review Checklist (NEI 0330, "Interface Reviews and Evaluations") is used to determine the appropriate disciplines required to perform this review. Independent design verification (NEI 0361, "Design Verification") is required for temporary modifications with a safety-related or augmented quality classification. PAP 1402 requires the performance of a 10 CFR 50.59 review as part of the development of the temporary modification.

Freeze seals are another temporary plant condition that is governed by PAP 1402. However, additional controls are specified in Generic Mechanical Instruction (GMI) 0024, "Freeze Seals." The GMI specifies that prior to implementation of a freeze seal, the freeze seal location, method, backfilling, and scope of work to be performed is submitted to engineering. Engineering reviews the information in accordance with Mechanical Desk Guide, MDS-S-001, "Freeze Seals", and provides additional information, such as, nondestructive examination requirements, safety plans, and 10 CFR 50.59 reviews. After completion of the engineering review activities, the GMI provides the controls needed to install and remove the freeze seal.

PAP 1402 also governs the installation of leak sealant devices. Additional controls for this temporary plant condition are contained in GMI 0095, "Instructions for the Use and Control of On-Line Leak Sealing." The GMI requires engineering review and approval of the leak sealant scope of work. In accordance with Mechanical Design Guide, MDS-MECH-005, "Leak Sealants", engineering evaluates parameters on items such as injection and system pressure, weight, and seismic effects. A 10 CFR 50.59 review is performed for the leak sealant activity. After completion of the engineering review activities, the GMI provides the controls needed to install the leak sealant device.

Temporary changes can be implemented to plant SSCs which do not receive the evaluations and reviews delineated in PAP 1402. In these instances, other specific actions are instituted to assure positive controls on the installation

and removal of the temporary change. Some of these instances/actions are provided below:

- Installation/removal of a temporary change while working under a troubleshooting log, or
- Utilization of temporary leads and jumpers for repetitive instrument calibrations performed under a work order or repetitive task.

Working under a troubleshooting log is further governed by PAP 0905, "Work Order Process", and requires the notification and authorization of the Control Room Unit Supervisor for installation/removal of the temporary change. In addition, personnel remain in the work area until the item is restored to normal configuration and documentation of correct installation/removal is performed and verified.

Utilizing temporary leads and jumpers as noted above, requires the Control Room Unit Supervisor to be notified prior to installation/removal of the temporary change. If a safety-related system is affected by the temporary change, the system is declared "inoperable" by the Unit Supervisor. In addition, calibration personnel remain in the immediate work area until the item is restored to normal configuration and independent verification is performed for the installation/removal. The temporary change cannot provide a change to the method of operating the system/component and cannot remain installed for longer than one (1) shift.

Internal Inspections/Assessments

A sampling of Quality Assurance audits associated with configuration control were reviewed. The scope of the audits reviewed included, but were not limited to:

- configuration control,
- setpoint change control,
- configuration database accuracy, and
- accuracy of documentation (e.g. procedures, drawings, USAR) associated with implemented design changes.

The review concluded that the audits found the configuration control processes and the implementation of the processes adequate.

Performance

The Preoperational Test phase at PNPP was conducted on an integrated system or subsystem basis to verify that the systems were capable of operating in a safe and efficient manner compatible with the system design bases. To the extent practical, the objectives of the preoperational test phase were to:

- verify the adequacy of the plant design,
- verify that plant construction was performed in accordance with the design,
- demonstrate proper system/component response to postulated accidents and malfunctions,
- confirm the adequacy of plant operating and emergency procedures, and

- familiarize the plant operating, technical, and maintenance personnel with the plant.

Testing or acceptance criteria, to verify design specification requirements, were incorporated into the test specifications. The Test Program Manual established the methods for preparing, reviewing, and controlling Initial Check and Run-in (IC&R) test procedures, preoperational test procedures, acceptance test procedures and test administrative procedures. Test procedures were prepared using information and requirements in documents such as NRC regulatory guides, General Electric test specifications, Gilbert/Commonwealth's design and test specifications, Final Safety Analysis Report (FSAR), technical manuals, and applicable codes and standards. Test specifications were used as the primary source for preparing system test procedures. Prior to release for performance of a test, preoperational and acceptance test procedures were reviewed and revised, as necessary, to incorporate the latest design information available.

In June 1985, a project plan, SPP 1102, "Test Procedure Assurance Program Plan", was implemented to assure that test commitments were identified and satisfied, and that preoperational and acceptance tests were technically adequate. The final report for SPP 1102 concluded that:

- testing commitments were identified and incorporated into the preoperational test procedures, and
- test procedures used during the Preoperational Test phase were adequate to demonstrate, to the extent practical, the capability of structures, systems, and components to perform satisfactorily when in service.

Preoperational and acceptance test results were reviewed and approved by the plant testing organization. Preoperational and acceptance test results were then submitted as part of the final system acceptance to the operational staff. When a system was accepted by the operational staff, system control was then administered through the use administrative procedures and instructions contained in the PNPP Operations Manual.

The primary SSC testing program is identified in PAP 1105, "Surveillance Test Control", which provides for the scheduling, performance, approval, and retention of all surveillance testing performed by Technical Specification Surveillance Instructions (SVI), Inservice Inspection Instructions (ISI), and Periodic Test Instructions (PTI). A matrix is included in this procedure which identifies the following:

- cross-reference between the Technical Specification Surveillance Requirement and the implementing instruction,
- type of test activity,
- test frequency/interval, and
- work group responsible for surveillance/test performance.

The primary purpose of this matrix is to assure that each Technical Specification Surveillance Requirement is identified and procedurally addressed. PAP 1105 also requires the operations and systems engineering groups review of the Surveillance Instruction results for acceptability.

Complimentary to the Surveillance Test Program are a number of other testing programs which verify that SSCs will perform in accordance with the design bases:

1. A Post Modification and Maintenance Test Program is in place to assure appropriate testing is performed following maintenance or modification activities. Maintenance activities and modifications are reviewed, and testing is specified in accordance with PAP 1123, "Pre-Maintenance and Post Maintenance/Modification Test Program." A Post Maintenance Test Manual (P[illegible] 036) was developed to help identify applicable testing requirements for SSCs following maintenance or modification. Utilization of this manual helps assure that the appropriate testing is performed to confirm continued SSC function per design requirements.

2. A Motor Operated Valve (MOV) Testing and Surveillance Program was instituted in order to assure the ability of safety-related MOVs to operate under design basis conditions. This program was established in response to NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance." The program is described in PAP 1116, "Motor Operated Valve Testing and Surveillance Program." The program specifies the guidelines and controls for maintaining an effective program governing work, setpoints, testing, and performance trending on safety-related MOVs.

3. The ASME Code Section XI In-Service Inspection (ISI) and In-Service Test (IST) Programs are administered in accordance with PAP 1001, "Inservice Examination Program"; PAP 1101, "Inservice Testing of Pumps and Valves"; the In-Service Examination Program (ISEP); and the In-Service Testing Program (ISTP). These programs control the performance of inspections, examinations and tests which assure the structural and pressure boundary integrity of safety-related components, and the operational readiness of safety-related pumps and valves in accordance with the requirements of the ASME Code Section XI and the PNPP Technical Specifications.

4. PAP 1115, "Snubber Augmented Visual Inservice Inspection/Examination and Functional Testing Program", describes the snubber inspection, examination, and functional test program in place at PNPP. This procedure applies to safety-related and/or Seismic Category I snubbers on systems affecting the safe operation of the plant. The functional testing described in the procedure satisfy the requirements of the Technical Specifications and ASME Code Section XI (IWF-5000). The visual inspection/examination and service life monitoring requirements contained in PAP 1115 satisfy the requirements of the Operations Requirements Manual. The visual inspection/examination and functional testing of nonsafety, nonseismic snubbers is also delineated in this procedure.

Other programs have been established to enhance plant reliability. Examples are:

1. A Pipe Wall Thickness Monitoring (PWTM) Program has been developed in accordance with NRC Information Notice 87-01, "Thinning of Pipe Wall in Nuclear Power Plants." The PWTM Program is implemented through PAP 1002, "Pipe Wall Thickness Monitoring Program." The program is designed to provide adequate prediction and monitoring of piping such that wall thinning below design allowables does not occur in systems subject to significant flow accelerated corrosion.

2. System walkdowns are performed by engineering personnel utilizing the System Engineering Handbook as guidance. The guidance states that systems in operation should receive a monthly walkdown and systems in standby should receive a walkdown at least every other surveillance interval. The following are examples of the types of items that are reviewed:

   - Verify the system is being operated in accordance with the approved operating procedures.
   - Verify operating parameters are in accordance with the approved operating procedures.
   - Review the applicable Updated Final Safety Analysis Report operating description.
   - Determine if new deficiencies exist and if older ones may have degraded.

3. System Health and Status Reports are performed by engineering personnel in accordance with guidance contained in the System Engineering Handbook. These reports assess overall system condition, evaluate system performance against goals, provide management with a specific critical assessment of system performance and status, and provide an overview of trended performance data. The System Health and Status Reports are reviewed by a management oversight group, the System Engineering Review Board.

4. A Check Valve Reliability Program was established in response to concerns identified in Significant Operating Experience Report (SOER) 86-03, "Check Valve Failures or Degradation." This program prioritizes check valve inspections and testing based on functions required and system modeling. Periodic inspections are conducted in accordance with the Preventative Maintenance (PM) program and results are fed back into the PM program for monitoring reliability.

Internal Inspections/Assessments

Attachment 2 provided a brief description of various inspections/assessments that have occurred at the PNPP over the past several years. The following provides some additional information on topics specifically relevant to this section.

1. The System Operations and Test Review (SOTR) Program, conducted in 1994, was implemented to perform a direct, deliberate confirmation of system and equipment readiness and functionality for the High Pressure Core Spray (HPCS) System, the Off-Gas System, and the Steam Bypass and Turbine Control System. The SOTR Program included steps to determine if shortcomings observed in the implementation of various control processes (e.g. design control, preventative maintenance, testing) have adversely affected the functional adequacy of plant systems and equipment. A partial listing of the program results are as follows:

   - supplemental testing performed on the three (3) systems demonstrated that the system functional requirements have not been impacted by SOTR identified control process shortcomings,
   - the control process shortcomings are considered generic,

- the 3 systems were confirmed to function as required, and
- maintenance and periodic testing associated with the 3 systems is considered adequate to assure continued safe, reliable operation.

Although control process shortcomings were identified, the SOTR Program concluded that the HPCS System was capable of reliable operation and satisfying the requirements of the Technical Specifications. The evaluations of the Off-Gas System and the Turbine Control and Steam Bypass System indicated that both systems are capable of reliable operations.

2. The Systems Based Instrumentation and Control Inspection (SBICI), the Independent Safety Engineering Group's (ISEG) "Setpoint Applications" assessment and the Emergency Service Water System Operational Performance Inspection (ESWOPI) (Items 6, 7, and 8 of Attachment 2, respectively), each performed some review of how SSCs are configured, tested and operated. Some concerns and weaknesses were identified in the SBICI and the Setpoint Applications assessment, but none were of significance with respect to nonconformance to design basis envelope requirements and/or affecting SSC operability. Issues identified by the ESWOPI are still being evaluated, as part of the PNPP Corrective Action Program.

(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.**

The response to this request provides a description of the processes in place at PNPP which ensure the prompt identification of conditions adverse to quality and mechanisms for implementing corrective actions, actions to determine extent of problems, prevention of recurrence of problems and prompt reporting to the NRC. The two key programs at Perry Nuclear Power Plant (PNPP) are the Corrective Action Program, which includes reporting to the NRC and PNPP Ombudsman Program, both of which are described in detail below.

Overall, the PNPP Corrective Action Program satisfies the key elements of problem identification, root cause analysis, extent-of-condition determination, corrective action to prevent recurrence, and reportability to the NRC. Making operability determinations, using guidance from Generic Letter 91-18, is integral to the process.

Corrective Action Program

The PNPP Corrective Action Program is used to identify problems, identify their causes and to implement actions to correct the problems. The corrective action program implements the requirements of 10 CFR 50, Appendix B, Criteria XV, "Nonconforming Materials, Parts, or Components", and XVI, "Corrective Action." The PNPP program is described in Plant Administrative Procedure (PAP) 1608, "Corrective Action Program."

The Potential Issue Form (PIF) is the document which is used to identify, document, investigate, track, and correct potential problems raised by the plant staff or others. There is no prescribed minimum threshold for generating a PIF. If an activity does not meet an individual's expectations, then a PIF should be generated. Any individual on-site may initiate a PIF. Initiation of a PIF requires no supervisory approval.

Once written, the PIF is submitted to the PNPP Control Room. The Control Room Shift Supervisor reviews the PIF to determine whether the issue affects the operability of any plant system or component, the issue is reportable, or the issue impacts the Technical Specifications. PAP 1608 provides the necessary guidance to process the PIF should one of these conditions exist. Regulatory Affairs personnel perform a regulatory reportability review on each PIF shortly after it is initiated. This review assures that initial notification requirements have been satisfied and determines if additional reporting is required. PIFs are also reviewed by Maintenance Rule personnel to determine if the condition impacts the PNPP Maintenance Rule Program.

For conditions determined to be potentially reportable under 10 CFR 21, a checklist contained in PAP 1608 provides guidance to aid in analyzing the condition with respect to the reportability determination.

If the condition potentially affects the operability of any plant System, Structure, or Component (SSC), PAP 1608 provides guidance with respect to making the operability decision. The guidance contained in PAP 1608 is derived from

many sources, including Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability." The guidance requires the evaluator of the condition to define the importance of the SSC's function to safety (by researching design basis documentation, as necessary) and to assess the impact of the identified problem on the SSC's ability to function as intended by its design.

The guidelines associated with the performance of any notification with respect to 10 CFR 50.20, 10 CFR 50.72, or 10 CFR 50.73 are contained in PAP 1604, "Reports Management." These guidelines are based, in part, on NUREG 1022, "Event Reporting Guidelines - 10 CFR 50.72 and 10 CFR 50.73."

The Corrective Action Program requires that each PIF be categorized. There are four categories with Category 1 being the most significant and Category 4 being the least significant. Category 1 and 2 PIFs require root cause analyses. Actions that are taken to correct the conditions are called "Corrective Actions To Prevent Recurrence (CATPR)." PAP 1608 provides guidance on how to perform a root cause analysis. Category 3 PIFs, being of lesser significance, require apparent cause analyses. Actions that are taken to correct the conditions are called "Potential Issue Form Remedial Actions (PIFRA)." However, depending on the nature of the corrective actions, corrective actions for Category 3 PIFs may be assigned as CATPRs. Category 4 PIFs are of low significance and are typically used for trending based on the condition being corrected through normal work practices. An extent of condition evaluation is conducted for Category 1, 2, and 3 PIFs that have generic implications of like or similar occurrences at PNPP or elsewhere in the nuclear industry. For nonconforming conditions, at a minimum, a 10 CFR 50.59 Applicability Check is performed in accordance with PAP 0305, "Safety Evaluations", for every "use-as-is" and "repair" disposition. (Refer to Attachment 1 for a description of the PNPP 10 CFR 50.59 Program). For PIFs requiring root cause analyses, the analyses are performed by personnel who are trained and qualified to perform the root cause analyses.

The PIF is assigned to the work group that is most directly familiar and responsible for the condition. Once the investigation is complete, it is reviewed and approved by the work group's supervisor or manager. Category 1 and 2, and selected Category 3 PIFs, require a secondary management review by the Corrective Action Review Board (CARB) which is comprised of supervisors and managers from various on-site work groups. The function of CARB is to assure the adequacy of PIF investigations and of corrective action identification and implementation.

PIFs may not be closed until every corrective action has been verified as complete. Corrective actions must be approved and verified by the responsible work group's supervision and management.

PIFs are electronically tracked from initiation to closure. The tracking system is accessible to site personnel. Each PIF is assigned several codes to aid in trend evaluation. A PIF database exists which can be used to obtain trend information. Part of the Corrective Action Program requires work groups to periodically perform trend analyses on PIFs that were assigned to the work group for investigation. In addition, the Corrective Action Administrator is required to perform periodic analyses of PIFs to look for collective significance or potential adverse trends.

PNPP Ombudsman Program

The PNPP Ombudsman Program provides another method for plant personnel to report perceived nuclear safety or quality concerns. The Ombudsman Program is described in PAP 0217, "Perry Ombudsman Program." The Ombudsman reports to the Vice President - Nuclear, Perry. The program requires that identified concerns be addressed by plant management not directly involved with a concern and allows escalation of the concern to the highest management level necessary to achieve resolution. The program applies to all PNPP personnel, including contractors and consultants.

Potential concerns are received from concerned individuals either by direct contact with the Ombudsman or via the telephone or mail. Site exit interviews, at the time of termination, resignation, or contract completion, are conducted for as many individuals as possible. Reported concerns are handled with the highest degree of confidentiality possible given the nature and extent of the concern.

The Ombudsman Program encourages individuals to attempt to resolve their concerns through the normal plant management and quality assurance channels by using PIFs, work requests, or modification requests, as appropriate. However, the Ombudsman remains available to all individuals who choose, for whatever reason, not to pursue the normal problem resolution channels. When a concern is received by the Ombudsman, a determination is made regarding the plant or corporate organization best suited to investigate the concern. The investigating organization determines the validity of the concern and makes a judgment as to whether the concern should be transferred to another administrative process or corrective action process for resolution. The Ombudsman Program is therefore not separate from the PNPP Corrective Action Program, but merely serves as an alternate entry point into the formal process. In all cases, the actions required for resolution of the concern are determined, documented, and tracked to completion. The individual who originally provided the concern, if not done anonymously, is apprised of the outcome of the investigation. The Ombudsman periodically reports to the Vice President - Nuclear, Perry on the status of concerns being handled by the Ombudsman Program.

Oversight of the Corrective Action Program is governed by the Quality Assurance audit and surveillance function.

(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.**

The response to this request will provide an assessment of the effectiveness of the plant processes and programs which concludes the configuration of the Perry Nuclear Power Plant (PNPP) is consistent with the design bases. The processes/programs evaluated are the design and configuration control processes, the Updated Final Safety Analysis Report (USAR) Program, the 10 CFR 50.59 Program, and the Corrective Action Program.

In general, PNPP's programs are effective in assuring that Systems, Structures, and Components (SSC) configuration is consistent with the design bases requirements. Weaknesses identified in this attachment and within Attachments 1 through 4, although not adversely impacting SSC design function or operability, do not meet PNPP management's expectations. Accordingly, additional initiatives will be pursued to improve these areas, as appropriate. PNPP will continue to evaluate the design and configuration control processes through the performance of various types of inspections and assessments to assure continued improvement in these areas.

Design and Configuration Control

Background

The requirements of the design control process have remained consistent through the construction and operational phases of the PNPP. The design control program is based on Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (Revision 2, June 1976), and ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants."

During the construction phase, design and design review were the responsibility of Gilbert/Commonwealth, Incorporated as the Architect Engineer (AE) and General Electric as the Nuclear Steam Supply System Supplier (NSSS), subject to PNPP oversight. Both organizations were required to have written design control procedures which met the requirements of ANSI N45.2.11 in accordance with the Cleveland Electric Illuminating Company Nuclear Quality Assurance Program.

The operations phase PNPP design control program was implemented in 1985. The overall program included the control of design activities performed by or for, the PNPP engineering organization. The design control program, then and now, includes control of each individual design change/modification/addition, and the resulting as-designed, as-built, as-tested documentation.

The construction phase and the initial PNPP programs contained critical elements that are maintained in the current PNPP design control program. Some of these critical elements include, but are not limited to:

- Design Input Records,
- Calculation Preparation,

- Design Verification,
- Drawing Change Notices,
- Design Configuration Management,
- Training Requirements,
- Plant Procedures/Instruction Review,
- Final/Updated Safety Analysis Report,
- Field Change Requests, and
- Post-installation Testing.

An additional tie between the construction and operations phases was realized by:

- maintaining Gilbert/Commonwealth as the primary continuing services A/E through mid-1991 to address more complex design issues,
- obtaining and maintaining the Balance of Plant (BOP) design documentation records from Gilbert/Commonwealth,
- maintaining a staff augmentation constituent from Gilbert/Commonwealth, and
- maintaining General Electric to provide for the engineering evaluation of plant design changes that could impact the NSSS design bases.

Attachment 1 provided a general description of the design and configuration control processes currently in place at PNPP. Attachments 2 and 3 provided more detailed information on the processes which assure that Systems, Structures, and Components (SSC) are built, tested, and operated in conformance with design bases requirements. It is noteworthy that some key programmatic elements (primarily ANSI N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants") were and continue to be common and basic to the PNPP design and configuration control processes.

Current Status

The effectiveness of PNPP's design and configuration control processes have been measured since commencement of commercial operation through vertical slice assessments such as the Safety System Functional Assessments (SSFA), and integrated self-assessments, such as NRC Engineering and Technical Support style reviews. Such reviews allowed for flexibility in the assessment of both compliance and performance aspects. Attachments 2 and 3 provide summary descriptions of a number of these assessments. The following provides additional information on assessments/evaluations relevant to the effectiveness of the PNPP design and configuration control processes.

1. As a result of the NRC Engineering and Technical Support Inspection (ETSI) conducted in May-June 1994 (Inspection Report 50-440/94010), an aggressive effort to re-evaluate Design Change Packages (DCPs) worked during the PNPP Fourth Refueling Outage (RFO4) was implemented. Cross-discipline teams of senior PNPP personnel were mobilized to perform the re-evaluations and to generate design reports to document conclusions. The main issues reviewed were:

    - whether any Unreviewed Safety Questions (USQs) existed,
    - whether additional hardware changes were required, or
    - whether significant configuration control problems existed.

Results of the re-evaluation concluded that, beyond the DCPs noted in the NRC inspection report, no other DCPs had significant problems.

The re-evaluation did identify areas where the design change process could be improved. The re-evaluation did not specifically look at the user-friendliness or ease of retrievability of design basis information.

The re-evaluation effort of RF04 DCPs described above was then reviewed by a group of senior engineering personnel with extensive industry experience outside of PNPP, called the Oversight Review Committee (ORC). ORC's responsibilities included review/approval of the design reports generated by the re-evaluation effort. After comment resolution, ORC concurred with the design reports and overall conclusions of DCP adequacy. The ORC also agreed that the design change process needed a major upgrade and provided recommendations.

This re-evaluation concluded that DCPs with major design/design output documentation problems were isolated concerns. Further, the concerns were identified via testing prior to the affected system's return to service/operable status. The majority of the identified problems by this re-evaluation involved weaknesses in design documentation supporting the design output documents (e.g. calculation detail, system interaction consideration/documentation, design verification rigor). The overall design change process was evaluated as needing improvement.

During the balance of 1994 and early 1995, a major re-engineering of the design change process was undertaken. Process changes went into effect on February 1, 1995. The primary process improvements included mandatory design reports for major modifications, establishment of project teams for DCPs, conceptual design reports, multi-discipline reviews/approvals, and the creation of a DCP desk guide. Based upon subsequent assessments, these process improvements appear to be effective.

2. In June 1995, the Independent Safety Engineering Group (ISEG) conducted a Design Engineering Assessment and Review (DEAR) on design bases configuration. The primary focus of the DEAR was to gauge PNPP's overall design basis configuration from a user access and utilization viewpoint by reviewing a small cross-section of design basis documentation. The results of this assessment will be described in Attachment 6. As a secondary focus, design basis documentation quality and process controls were reviewed. Findings in this area were similar to those described in Item 1 above and Item 6 in Attachment 2 (e.g. weaknesses were identified in calculation detail and interfaces, design input completeness). No System, Structure, or Component (SSC) operability concerns were identified.

3. In October 1995, an Engineering and Technical Support Inspection (ETSI) was conducted in accordance with NRC Inspection Procedure 37550, "Engineering." This inspection, which also served as a design control audit, assessed the effectiveness of engineering to perform routine and reactive plant activities, including resolution of technical issues and problems. A major portion of the inspection was the review of design basis documentation used in support of the PNPP Fifth Refueling Outage

(RFO5) DCPs. The inspection concluded engineering performance was adequate. A strength was observed in the recently upgraded design change process and improvements noted in 10 CFR 50.59 Safety Evaluations. No SSC operability concerns were found. Weaknesses, however, were identified. The weaknesses included but were not limited to:

- programmatic control of design guides,
- improper engineering evaluation of partially implemented design changes,
- lack of design verification, as appropriate, within the Temporary Modification Program,
- lack of technical detail, in cases, in documentation used in support of design changes, and
- owner review of contractor engineering deliverables not consistently rigorous.

Although such issues were identified, none questioned the ability of the SSCs (associated with the DCPs reviewed) to perform their intended design bases function. For the issues identified, procedure/instruction changes, and administrative controls were implemented to effect improvement. In addition, a primary improvement involved the focused training of engineering personnel in 1996 in the areas of calculations, design interfaces, design verifications, and design bases.

4. The Engineering Assessment Review Team (EART) was established in December, 1995. This team is comprised of engineering, quality assurance, and ISEG personnel. The team reviews and classifies corrective action and other assessment documents involving engineering activities to determine significance, trending, and the need for additional corrective action or improvement initiatives. In addition, the EART provides recommendations for additional assessments and general engineering oversight in areas found to exhibit declining performance trends.

   The first review performed by EART involved an overview of the SBICI, DEAR, and ETSI inspection/assessments described in Item 6 in Attachment 2 and Items 2 and 3, above. The main objective was to review the individual assessment findings in a global sense and determine if individual corrective actions were adequate. EART concluded that no major design related issues existed, and that the extent-of-condition reviews from the individual assessments were adequate. However, several recommendations with respect to procedural enhancements (e.g. the calculation process) and future self-assessments (e.g. setpoint applications and surveillance instruction interfaces) were made and are being implemented.

5. In June 1996, a self-assessment of the Fire Protection Program was performed. The objectives of the self-assessment were to assess the Perry Nuclear Engineering Department's (PNED) abilities in implementing the PNPP Fire Protection design requirements and commitments and in approving, assigning, and tracking of fire protection tasks. In addition, an assessment was made of PNEDs design change program as it relates to fire protection. The assessment found that PNED is and has been active in the process of identifying areas of concern and implementing measures to correct them.

Strengths noted in the assessment include:

- The process for ensuring fire protection reviews are performed to evaluate a modification's impact on fire protection features and/or safe shutdown capability,
- The ongoing Thermo-Lag Reduction Analysis, with the goal to eliminate all Thermo-Lag,
- Safe Shutdown Methodology relies on operators' actions with procedures that they work with routinely, and
- Review of emerging issues as identified by the NRC.

No SSC operability issues were identified. However, a weakness was identified regarding the accessibility of design bases documentation.

Overall, the assessments have confirmed the adequacy of PNPP's design, and PNPP's design and configuration control processes to provide reasonable assurance of SSC conformance to design bases requirements. However, the assessments have identified some weaknesses in the area of process implementation including inconsistent quality of design bases documentation and ease of documentation retrievability.

Several improvement initiatives concerning design bases documentation quality (e.g. calculation detail, design input completeness, design verification rigor) have been implemented. These include:

- Focused training for engineering personnel in the areas of calculations, interfaces, design reviews, and design bases.
- Establishment of engineering policies in the areas of calculations, safety evaluations, and design bases documentation.
- Establishment of an Engineering Assessment Review Team (EART) with cross-departmental representation, to overview engineering issues/trends and make recommendations for improvement.

The Configuration Management Improvement Program (CMIP) was another initiative with a purpose of enhancing the implementation of the design and configuration control processes. Some of the objectives of the CMIP are:

- Provide definitions for configuration management and configuration related terms applicable to PNPP.
- Identify the SSCs that are subject to the policies and procedures of configuration management.
- Identify the PNPP document types, databases, and computer applications that contain configuration information relating to the selected SSC.
- Identify the primary configuration control processes in use at PNPP.
- Develop a desk guide to enhance the ability of an engineer to retrieve design bases information for the SSCs covered by the configuration management processes.

Concerns associated with design bases retrievability will be addressed in Attachment 6.

The effectiveness of the implementation of the design and configuration control

processes at PNPP will continue to be evaluated through a pattern of SSFA type assessments (typically using NRC inspection modules) presently planned for the foreseeable future. In addition, the results of these assessments will be evaluated for common themes and trends by the EART to provide senior management feedback on emerging engineering issues and to recommend improvement initiatives, as appropriate.

Updated Final Safety Analysis Report

The PNPP Updated Final Safety Analysis Report (USAR), the administrative controls for the USAR, and the USAR oversight processes were described in Attachment 1 of this letter. This section will describe the results of the oversight processes which evaluated the effectiveness of the controls associated with the USAR.

Background

The initial PNPP Final Safety Analysis Report (FSAR) was submitted to the NRC in September 1980. It was amended twenty-five times between initial submittal and the final amendment which was submitted in June 1986. In 1984-85, PNPP undertook a project to validate the accuracy and technical correctness of the FSAR. The project consisted of a team of individuals from Gilbert/Commonwealth (the Architect Engineer), General Electric (the Nuclear Steam System Supplier), and PNPP personnel. As part of the validation process, source documents were required to be listed which provided the basis for the various FSAR sections. The discrepancies that were identified were resolved by individuals from the three organizations. If FSAR changes were warranted, the changes were incorporated into the last several FSAR amendments. This validation provided PNPP with reasonable assurance that the FSAR was accurate and did reflect to the plant's design bases.

The FSAR provided the basis for the PNPP USAR. The initial PNPP USAR was developed in 1987-88, with an initial submittal to the NRC in March 1988. The primary purpose of the USAR development program was to submit the USAR in accordance with the requirements of 10 CFR 50.71(e), to incorporate any changes that may have occurred between the last FSAR amendment and the USAR submittal, and to improve the clarity of the USAR, where needed. The development program included review of the entire draft USAR by various on-site work groups, including the engineering and the operations organizations. One of the activities performed by this multi-work group review process was to verify the accuracy of the contents of the draft USAR. If changes were required, USAR Change Requests were submitted to make the changes. Use of the FSAR as the basis of the USAR combined with the USAR development program's accuracy review provided a measure of assurance that the initial USAR was accurate.

A sampling of Quality Assurance audits and surveillances conducted between 1986 and 1994 was reviewed with respect to the accuracy of the USAR and the effectiveness of the USAR change process. Of the audits and surveillances reviewed, limited information regarding the accuracy of the USAR or the effectiveness of its controls was obtained. One exception was Audit 94-09, "Design Control Program." It indicated various deficiencies within the USAR. The results of the audit were incorporated into the Corrective Action Program for documentation tracking, investigation, and correction. The deficiencies have been corrected.

Current Status

In late 1995 - early 1996, a self-assessment, titled the "Updated Safety Analysis Report Change Request Processing", was performed on the USAR change process. The self-assessment was designed to review the USAR Change Request Program including the administrative controls and processes to determine the effectiveness of the program. The results of the assessment indicated the USAR maintenance process was functioning adequately.

As a result of events at several other nuclear power plants and the issuance of NRC Information Notice 96-17, "Reactor Operation Inconsistent with the Updated Final Safety Analysis Report", an initiative was undertaken to validate a sample of the PNPP USAR. Four USAR subsections were selected for review. One subsection dealt with the basic design of mechanical subsystems and components. One subsection dealt with safety-related equipment and design features. The remaining two subsections dealt with the design and operation of nonsafety-related systems and components. The validation was performed by individuals from the on-site engineering and operations work groups. The basic guidelines for the validation was to determine the technical accuracy of the USAR subsections under review. Any discrepancy found was to be noted and categorized with respect to its significance. The discrepancies were to be evaluated for adverse trends and be corrected through the USAR Change Request process. If any adverse trends were indicated, actions would be developed to correct these trends. The project is still in progress. Most of the discrepancies found are considered to be clarifications or enhancements of existing information or typographical corrections. However, Potential Issue Forms (PIF) have been written on a number of discrepancies due to their significance. These PIFs are being documented, tracked, evaluated, and corrected through the Corrective Action Program.

As a result of IEN 96-17, Quality Assurance Surveillance 96-055, "Corrective Action Evaluation of USAR", was performed to evaluate USAR-related corrective action documents that were written between 1995 and 1996 and the actions taken to assure that the plant remains consistent with the USAR. The results of the surveillance indicated that the site organization occasionally fails to update the USAR, resulting in the USAR not being accurate, and that the site organization occasionally fails to follow USAR requirements in the performance of some plant activities. The results of this surveillance are being documented, tracked, and evaluated in accordance with the PNPP Corrective Action Program. The evaluation of these issues is still in progress.

Based upon the above information, there is reasonable assurance that the discrepancies found in the PNPP USAR are minor. However, the audits/assessments described above indicate that compliance with the controls associated with maintaining the USAR need improvement. The completion of the 1996 sample USAR validation, as well as an evaluation of the corrective action documents listed in Surveillance 96-055, should assist in determining where program compliance improvements should be made. The results of these activities, and any future actions made with respect to the PNPP USAR, will be submitted to the NRC, if warranted, in separate correspondence.

10 CFR 50.59 Program

Background

The PNPP implemented the 10 CFR 50.59 process in 1983, three (3) years prior to the receipt of the PNPP Operating License. Due to the structure of the on-site organization at the time, two 50.59 procedures/instructions were in use. The Perry Plant Technical and Perry Plant Operations Departments followed Perry Administrative Procedure (PAP) 0305, "Safety Evaluations." The Nuclear Engineering Department followed Nuclear Engineering Instruction (NEI) 0332, "Safety Evaluations." Although two documents were in use, the guidance was essentially the same.

In 1986, training was implemented to enhance the ability of the individuals performing 50.59 evaluations. In 1989, industry guidance on the performance of 50.59 evaluations was developed and incorporated into PAP 0305. Further, as part of this PAP 0305 change, NEI 0332 was then canceled bringing the entire on-site organization under one program for the performance of 50.59 activities. With the implementation of the revised procedure, the 50.59 training program was also revised. The changes to the training program included the incorporation of a Updated Final Safety Analysis Report (USAR) module and an Accident Analysis module. These modules were added to enhance the knowledge level of the 50.59 evaluators.

In 1993, a computerized USAR database came on-line. This database allows individuals to perform word searches on the USAR to aid in the performance of 50.59 evaluations.

A sampling of Quality Assurance audits and surveillances conducted between 1986 and 1994 were reviewed to assess the adequacy of the 50.59 program. The results of the audits/surveillances, in general, indicated that the 50.59 evaluations were adequate. Of the audits/surveillances evaluated, no Unreviewed Safety Questions (USQ) were identified. However, several of the audits/surveillances identified instances where personnel did not follow the guidance contained in PAP 0305.

Current Status

In 1994 and again in 1995, the Director, Perry Nuclear Engineering Department (PNED) chartered a team of experienced individuals to review the 50.59 evaluations associated with various engineering activities. The team sampled various engineering activities performed over a given time period. The results of both reviews indicated that while no USQs were found, the quality of some 50.59 evaluations did not meet the expectations of the team. Furthermore, both reviews indicated that the guidance contained in PAP 0305 may not have always been followed (e.g. the administrative process was not properly followed). To correct these deficiencies, the Director, PNED, issued several policy statements to engineering personnel regarding the quality of 50.59 evaluations. Key engineering department personnel received enhanced 50.59 training to help improve the 50.59 evaluation quality.

The PNPP off-site review group, the Company Nuclear Review Board (CNRB), is chartered to review 50.59 Safety Evaluations and assure that no USQs exist. Personnel interviewed with respect to this function indicate that no USQs have been found.

Based upon the above information, the 50.59 program is effective. However, program implementation could be improved. Evaluation of the program will continue through the Quality Assurance audit and surveillance function. 10 CFR 50.59 Program changes, including enhancements, will be made based upon the results of these audits and surveillances.

Corrective Action Program

Background

The initial PNPP Corrective Action Program was comprised of four separate and independent programs. The programs were described in the following procedures/instructions: Plant Administrative Procedure (PAP) 0124, "Radiological Awareness Reporting and Occurrence Program"; PAP 0606, "Condition Reports"; PAP 1501, "Identification and Control of Deficient Items"; and Nuclear Quality Instruction (NQI) 1601, "Corrective Action." These four programs assured compliance with 10 CFR 50, Appendix B, Criterion XV and XVI.

A sampling of Quality Assurance audits conducted between 1986 and 1994 were reviewed with respect to the adequacy of the PNPP Corrective Action Program. These audits indicated that the Corrective Action Program had a number of weaknesses including timeliness of corrective action, adequacy of corrective action to prevent recurrence, inconsistent quality in event investigation, and trending. Some of the weaknesses were corrected.

Current Status

The Corrective Action Program was completely revised in 1994 as a result of commitments made in the Perry Course of Action. The previously referenced four programs were incorporated into a single corrective action program described in PAP 1608, "Corrective Action Program."

A sampling of Quality Assurance audits conducted between 1994 and 1996 were reviewed with respect to the adequacy of the "new" Corrective Action Program. One of the audits indicated that the Corrective Action Program is generally effective and improving. The most recent audit concluded that the Corrective Action Program is effective in identifying and resolving issues adverse to quality. However, the results of this review indicate that the timeliness of corrective action and the adequacy of corrective actions to prevent recurrence are items that may need further improvement.

A self-assessment of the "new" Corrective Action Program was performed in 1995. The self-assessment #013QCS95, "Perry Nuclear Assurance Department Corrective Action Program Self Assessment", did not provide a specific conclusion with respect to the overall effectiveness of the program. The self-assessment did make several recommendations for program enhancement which were subsequently incorporated into PAP 1608.

An independent assessment of the Corrective Action Program was performed in 1996. The assessment, "Independent Assessment of the Corrective Action Program for Centerior Electric's Perry Nuclear Power Plant", identified several areas that could use improvement. The areas incluue the timeliness of some corrective

actions, resolving recurring issues, root cause analysis, and trending. Many of the assessment's recommendations were incorporated into a revision to PAP 1608 which became effective in July 1996.

The issues described above are viewed as opportunities for continuing improvement of the Corrective Action Program. In its current form, the Corrective Action Program does bring an appropriate level of attention and review to design basis and configuration management issues. In this respect, the Corrective Action Program is currently providing an acceptable level of effectiveness in demonstrating that the configuration of the plant is consistent with the design basis. Evaluation of the adequacy and the implementation of the Corrective Action Program will continue through the Quality Assurance audit and surveillance function. Corrective Action Program changes, including enhancements, will continue to be made based upon the results of these audits and surveillances.

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

The Nuclear Regulatory Commission's (NRC) letter requested a response to one additional item beyond items (a) through (e). The request was to provide information relative to any design review or reconstitution program at the Perry Nuclear Power Plant (PNPP). PNPP's design bases process, in the aggregate, have been effective based upon the situations where engineering has been generally able to access the appropriate analyses and documents in response to addressing safety significant issues. Engineering has been able to obtain design bases information for use in the design change control process. The existence of an effective Corrective Action Program provides the mechanism where the extent of condition and root cause of an issue involving design bases information would receive a thorough investigation and implementation of corrective actions. The above information, coupled with PNPP's policy on Design Bases Information Reconstitution, formed the conclusion that Design Bases Documents (DBD) are not required at this time. The following paragraphs supports this conclusion.

Internal Assessments and Audits

It is recognized that design basis documentation accessibility/retrievability is important to adequately support an operating nuclear power plant in areas such as design change development and operability determinations. Multiple assessments at PNPP have evaluated this area, including:

1. A focus group was established in 1991 to analyze the applicability of NUMARC 90-12, "Design Basis Program Guidelines" to the PNPP. The group recognized the NRC's concerns regarding design basis reconstitution but viewed them as primarily focused towards earlier vintage plants (than PNPP) with less stringent documentation and quality assurance programs. In the opinion of the focus group, suitable documentation was available either through maintained records from the Architect Engineer (Gilbert/Commonwealth) or through on-going interface with the retained Nuclear Steam Supply System vendor (General Electric). It was recognized, however, that Design Basis Documents (DBD) could enhance overall efficiency of engineering personnel in performance of tasks utilizing design basis documentation. Therefore, the group received PNPP management approval to pursue a DBD pilot project. Following the NUMARC 90-12 guidelines, several draft DBDs of various format types were generated for a plant system and issued for management review in December 1992. Management evaluation of the DBD drafts concluded that resources would be best devoted to upgrading design documents on an on-going basis, rather than consolidating information in DBDs.

   An assessment of PNPP's configuration management processes was performed by an independent consultant in October 1993. The purpose of the assessment was to perform an overview evaluation of the processes with respect to industry guidelines and other common industry practices, and to

NOTE: *Section (f) designation applied by licensee.

provide recommendations for improvement. With respect to the need to embark on a design basis reconstitution program at PNPP, the consultant concluded that PNPP was in need of a document outlining the hierarchy of design bases information. It was concluded that good design bases documentation is available, but there were some observations of inconsistent control and use of the documentation (e.g. document revision levels not referenced in design inputs, calculation logs not kept updated). Recommendations were made for various process improvements. The assessment recommended a continuation of the DBD pilot project to assess the need for and to evaluate the cost-benefit relationship of DBDs.

2. In June 1995, the Independent Safety Engineering Group (ISEG) conducted a Design Engineering Assessment and Review (DEAR), entitled "Design Bases Configuration." This evaluation concluded that design bases documentation was not configured to promote user ease-of-access. It should also be noted that no issues of significance [i.e. potential impact upon any System, Structure, or Component (SSC) operability] were cited in the assessment due to the observed difficulty in documentation retrievability. Rather, the review concluded that this condition causes worker inefficiencies and could cause an increased potential for design-related human errors if design bases documentation was not rigorously researched.

3. In August 1995, as part of the Configuration Management Improvement Plan (CMIP), an assessment was performed to review and analyze the results of various Configuration Management (CM) assessments and other activities already performed at PNPP. The objective was to determine if additional CM improvement initiatives were warranted beyond those already identified in the CMIP. The period of time targeted was between January 1993 and May 1995. Numerous documents were reviewed. As part of this review, two categories considered were "design bases not available" and "design output documents not in agreement with the design bases." Assessment of these two areas concluded:

   - Very few cases were identified as design bases documentation not being available, and
   - Relatively few cases were identified where design output documents were not in agreement with design basis requirements, specific cases identified were relatively minor with respect to SSC function.

4. Other assessments, as described in Attachments 2 and 3, included evaluation of design bases documentation accessibility. These assessments, such as the System Operation and Test Review (SOTR) Program conducted in 1994 and the Fire Protection Self-assessment performed in 1995 made similar observations to those contained in Item 2 above. They recognized the difficulty, in several cases, of being able to quickly retrieve design basis information. However, none of the assessments cited any examples of SSC operability impacts due a lack of accessibility of design bases documentation.

5. A sampling of Quality Assurance audits related to design control were reviewed. The scope of the audits reviewed included, but were not limited to:

    - adequacy of design input,
    - completeness of calculations,
    - correct use and control of design output documents,
    - adequate design interfaces, and
    - adequate design verification.

    The results of this review indicated that the audits found the design processes and process implementation to be adequate.

Conclusion

In 1995, the decision was made that PNPP would not embark on a major design bases reconstitution program. This decision was based on assessments performed under a Configuration Management Improvement Program (CMIP) initiative that analyzed a multitude of observations from various assessments, corrective action documents, etc. The review concluded that there were no significant repetitive issues or trends relative to the availability of design basis information. Assessments subsequent to this decision have not provided any new data that would warrant a change in this position.

Supporting the decision that design bases reconstitution at PNPP is not warranted at this time are the following:

- Cases of inadequate design are isolated. Plant configuration continues to meet design basis requirements.
- Design basis documentation issues center more on overall documentation thoroughness not meeting expectations (e.g., calculation detail), rather than on a lack of understanding/maintenance of the design bases.
- Lack of retrievability of design basis documentation, in some cases, is primarily an issue of worker efficiency and familiarity at PNPP, not a concern for loss of design bases control.

Retrievability of design bases documentation has not manifested itself as a plant safety significant issue. Nor has this difficulty caused any noteworthy adverse impacts on the Perry Nuclear Engineering Department's ability to support the operating facility when responding to operability determinations, developing design changes, or handling day-to-day plant activities. The difficulty lies in the timeliness of retrievability which will be addressed in on-going improvement initiatives. PNPP's design bases issues can be characterized as weaknesses; however, they are not loss-of-control issues. Thus, a substantial reconstitution effort has not been considered warranted. Processes, as described in Attachment 1, are in place to control SSC configuration in the day-to-day functioning of the facility, to provide in-depth design reports associated with design changes, and to perform continuous assessment of SSC operability during investigation of potential issues as part of the Corrective Action Program.

Although a major design basis reconstitution effort has not been implemented by PNPP, improvement initiatives have been, and are being, implemented in the design bases area. As part of the Perry Course of Action, the CMIP implemented

recommendations regarding establishment of a Design Bases Hierarchy Desk Guide to enhance the engineers ability to locate and retrieve design bases documentation. The plant document types, data bases, and computer applications that contain configuration information are also described in this guide.

The Perry Plan for Excellence (PPE), PNPP's strategic business plan for the future, has initiatives for configuration management which include improving the standardization and control of PNPP calculations. This initiative is a continuation of activities implemented under the Configuration Management Improvement Program for improving retrievability of design basis information. It includes centralizing calculation information, standardizing and automating calculation logs/logging requirements, and establishing a central calculation reference library. Furthermore, efforts are underway to improve the availability and quality of PNPP vendor manuals and drawings by automating the technical information. This program involves the transfer of over 1400 vendor file folders containing 9000 vendor technical documents into a database for access by maintenance planners and technical support personnel. Project personnel are executing this effort under controlled processes which evaluate and resolve document legibility issues, and eliminate duplications and multiple revisions. This effort, coupled with PNPP's efforts under Generic Letter 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2 (Vendor Interface for Safety-Related Components)" will greatly improve the reliability and retrievability of vendor file information.

Perry Nuclear Engineering Department policies also recognize the potential difficulties involved in establishing design bases and set expectations for reconstitution of missing information. In addition, the policies outline expectations for the control and maintenance of documents carrying design basis information. The control processes outlined in Attachment 1 are designed to assure correctness of the information within the hierarchy of design basis documentation.

The effectiveness of the overall design and configuration control processes at PNPP will continue to be evaluated through a pattern of SSFA type assessments. This type of assessment is planned to be implemented at a frequency of one (1) SSFA per reactor operating cycle on risk significant systems/trains/functions within the scoping criteria of PAP 1125, "Monitoring the Effectiveness of Maintenance Program Plan" (Maintenance Rule). In addition, various integrated self-assessments such as NRC Engineering and Technical Support style reviews and Design Engineering Assessment and Review (DEAR) evaluations conducted by ISEG will continue throughout the foreseeable future. These type of assessments, in conjunction with the Corrective Action Program and oversight by the Engineering Assessment Review Team (EART), will be the mechanisms utilized to continue to confirm PNPP's operation within its design bases.

The existing Perry Configuration Management Improvement Program (CMIP) initiatives have been focused primarily in the Perry Nuclear Engineering Department. The processes presently included in the CMIP need to be integrated into other functional areas site-wide to further assure that the plant's physical and functional characteristics will be maintained in conformance with the design bases. PNPP will establish a site-wide administrative procedure and revise existing plant procedures/instructions to reflect the Configuration Management hierarchy. This commitment is intended to further assure that the

processes and programs utilized to operate, maintain, and performance monitor SSCs important to safety accurately reflect the supporting design bases. PNPP plans to utilize some of the guidelines contained in NUREG-1397, "An Assessment of Design Control Practices and Design Reconstitution Programs in the Nuclear Power Industry", and NUMARC 90-12 to further enhance this program.

Perry has recently created a Configuration Management Unit which will be tasked with the overall coordination of this activity. The plan for these procedure revisions and program enhancement activity will be transmitted to the NRC under separate cover letter by August 31, 1997.