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NewYorkPower Authority

February 7, 1997
JPN-97-005
IPN-97-016

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Washington, DC 20555

SUBJECT: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information

REFERENCE: NRC letter, James M. Taylor to Robert G. Schoenberger, dated October 9, 1996 regarding "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information"

Dear Sir:

Attached are two reports which respond to the NRC's October 9, 1996 request for information (Reference) regarding the adequacy and availability of design bases information. Attachment I provides the requested information for the Indian Point 3 Nuclear Power Plant. Attachment II is for the James A. FitzPatrick Nuclear Power Plant.

The Authority is aware of the importance of maintaining configuration control. Many initiatives and programs have been conducted since Indian Point 3 and FitzPatrick received their Operating Licenses. Each program or initiative was designed to address a specific aspect of plant design, operation or maintenance. Some of these programs were self-initiated, while others were developed at the NRC staff's request. Some were large in scope and resource intensive, such as the Design Basis Document Program. Other initiatives were smaller. As part of the Authority's commitment to excellence and continuous improvement, more initiatives like those already performed will be conducted in the future.

The Authority is confident that adherence to the processes described in these documents provides reasonable assurance that design bases requirements are being properly translated into design specifications, operating, maintenance and testing procedures and, that the configuration of structures, systems and components are consistent with the design bases.

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These processes are also designed to assure that when inconsistencies are found, they are evaluated and proper corrective actions are taken. Numerous efforts, including reviews, inspections, audits and walkdowns that have been conducted, and which will continue, further assure the consistency between the design basis, the plant and its operation. As an additional check, Authority management expects each employee to exhibit a questioning attitude and to raise questions to management's attention.

The two reports are each divided into five sections to address the area of information specified in the October 9, 1996 letter (sections a-e) and also includes a section discussing the Authority's design review/reconstitution program initiatives (section f). Both reports focus on the "design bases" as defined in 10 CFR 50.2 and amplified in footnote 4 of the NRC's October 9, 1996 letter. The reports are not all inclusive. Important programs, processes and on-going initiatives with strong ties to design bases information are highlighted in the two reports. Programs and initiatives only indirectly related to the design bases are not described.

The processes described in the reports are representative of the processes in-place when the report was written and are subject to change as needs develop. Changes to improve and clarify existing procedures are an on-going process.

A separate letter will be submitted detailing the Authority's commitments with regard to design basis information. The Authority will submit this letter, which will include plans and schedules for both Indian Point 3 and FitzPatrick, by March 10, 1997. This letter will include a description of initiatives to identify and correct any FSAR (Final Safety Analysis Report) noncompliances in accordance with the provisions of the Commission's amended policy statement (61 FR 54461, October 18, 1996) on enforcement actions associated with departures from the FSAR.

Attachment III summarizes the commitments made in this submittal. If you have any questions regarding this matter, please contact Ms. C. Faison, Director - Nuclear Licensing.

Sincerely,

Harry P. Salmon, Jr.
Chief Nuclear Officer, Acting

cc: next page

cc: Regional Administrator
U. S. Nuclear Regulatory Commission
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<u>List of Attachments</u>

I. Response to Request for Additional Information Pursuant to 10 CFR 5054(f) Regarding Adequacy and Availability of Design Bases Information, Indian Point 3 Nuclear Power Plant, dated February 7, 1997

II. Response to Request for Additional Information Pursuant to 10 CFR 5054(f) Regarding Adequacy and Availability of Design Bases Information, James A. FitzPatrick Nuclear Power Plant, dated February 7, 1997

III. Summary of Commitments

IV. Affidavit

Response to Request for Additional Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information for the Indian Point 3 Nuclear Power Plant

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

February 7, 1997

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EXECUTIVE SUMMARY

This report responds to the NRC's October 9, 1996 request for information (Reference 1) regarding the adequacy and availability of design bases information for the Indian Point 3 Nuclear Power Plant.

The information in this report was collected by a team composed of experienced Authority employees and a contractor. This team worked to identify the current processes, on-going initiatives and past programs that affect the design bases, the configuration control of the facility, and the engineering processes used. After identifying these, the team worked with individuals familiar with each topic to prepare a summary description of the process, initiative or program, focusing on how the information was used to confirm the adequacy and availability of design bases information. The team reviewed the descriptions provided and prepared descriptions to condense the information to be responsive to the information requested. The preparers were asked to verify in writing the accuracy and completeness of their information.

The Authority is confident that adherence to the processes described in these documents provides reasonable assurance that design bases requirements are being properly translated into design specifications, operating, maintenance and testing procedures and, that the configuration of structures, systems and components are consistent with the design bases.

INTRODUCTION

This report responds to the NRC's October 9, 1996 request for information regarding the adequacy and availability of design bases information for the Indian Point 3 Nuclear Power Plant.

The Authority is committed to maintaining configuration control. Many initiatives and programs have been conducted since IP3 received its Operating License. Some of these programs were self-initiated, while others were performed at the NRC staff's request. Some were large in scope and resource intensive, such as the Design Bases Document Program. Other initiatives were smaller. More initiatives like those already performed will be conducted in the future.

The report is not all inclusive. Important programs, processes and on-going initiatives with strong ties to design bases information are provided in the report. Programs and initiatives indirectly related to the design bases are not described.

The procedures described in the report are representative of the processes in-place when the report was written and are subject to change as needs develop. Changes to improve and clarify existing procedures are an on-going process. In addition, changes to titles and other organizational changes are currently being implemented.

DEFINITION OF DESIGN BASES

This report uses the definition of "design bases" in 10 CFR 50.2 and amplified in footnote 4 of the NRC's October 9, 1996 letter (Reference 1). Because "design bases" information is only a fraction of the information required to design, construct, license and operate Indian Point 3, other terms are used to represent this larger body of information.

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

BACKGROUND

Configuration Control of the plant is established and maintained through processes that control the as-built condition of Structures, Systems, and Components (SSCs) with respect to approved documents and data bases. These documents and data bases are developed using procedures that provide direction in assuring design bases requirements are properly translated into technical requirements for SSCs. This information is indexed and filed so that it is retrievable by plant personnel for use in operating, maintaining, and, if required, modifying the nuclear facility. Nuclear Administrative Policies (NuAPs) provide the policies and management expectations in those areas that maintain the configuration and design of the nuclear facility. The objectives of and the elements that constitute the Configuration Management Program are described in the Authority's Configuration Management Manual (CMM) procedure CMM 1.1.

Change processes are controlled by procedures in the Modification Control Manual (MCM). This manual contains the processes for implementing changes and modifications to plant SSCs including software changes and setpoint changes. Modification procedures integrate the functions of initiation, preparation (including design change), installation, testing, return to service, and the updating of documents and data bases. Specific requirements for the performance of tasks associated with modifications are contained in; (1) the Design Control Manual (DCM) for design requirements, (2) the Configuration Management Manual (CMM) for software requirements and document control, (3) Site Design Engineering procedures SED-ADs for site specific tasks, (4) Site Administrative Procedures (APs) for requesting and processing modifications using Problem Identification Description (PIDs) and Work Requests (WRs), and (5) the Engineering Standards Manual (ESM) for analyses such as breaker and fuse coordination, stress analysis of piping systems, and instrument loop accuracy and setpoint calculations.

ORGANIZATION

The Nuclear Generation Business Unit is under the direction of the Chief Nuclear Officer (CNO). The Nuclear Generation Business Unit contains five organizations consisting of the two nuclear facilities(Indian Point 3 (IP3) and James A. Fitzpatrick (JAF) Nuclear Power Plants), Appraisal and Compliance Services Department (Support Function reports to the Vice President, Appraisal and Compliance), Nuclear Engineering and Project Control Division, and Nuclear Business Operations Section.

Configuration and engineering processes are under the direction of the Vice President Engineering and Project Control (VPE-PC) who reports to the CNO. The VPE-PC controls the engineering and configuration processes through the Director of Nuclear Engineering located in the corporate office. Design and configuration control are the responsibility of the Directors of Design Engineering (two directors, one located at each of the nuclear facilities) and the Director of Engineering Support (located in the headquarters office) who report to the Director Nuclear Engineering. These directors establish the processes that control those tasks required to maintain the nuclear facility's design and configuration.

APPENDIX B TO 10 CFR 50

The information being provided is a description of engineering and configuration control processes. Included in detail below are those most relevant to ensuring the plant configuration and performance are consistent with the design bases. The IP3 Nuclear Quality Assurance Program is described in the IP3 FSAR and NYPA Quality Assurance Program Manual.

DESIGN CONTROL

Design control processes are performed in accordance with the procedures in the Design Control Manual (DCM). Guidance to the Authority's approach to Design Control is provided in procedure DCM-1 "Design Engineering Activities." This procedure gives an overall process for completing design changes to the nuclear facility and provides guidance on the approach to be used for resolving problems.

Procedure DCM-13 "Conduct of Engineering" establishes requirements for the administrative controls for the conduct of engineering activities to address technical, safety, design bases, environmental, codes and standards, and regulatory guidance. DCM -13 addresses the engineering/design process. It uses checklists to (1) identify items to be considered for design input including the design bases, (2) provide guidance for the performance of the design, and (3) identify items to be checked at design completion. Design changes for modifications using the MCM-3 process (major modifications), require that the checklists be completed and submitted as part of the design change package. Design changes for MCM-5 (minor modifications), require the procedure and checklist to be followed but the checklists do not need to be completed and the design information generated is included in the modification package. Design changes for MCM-14 (equivalencies and small design changes) do not require use of procedure DCM-13 unless directed by the Director of Design Engineering. Information for the design change is included in the Type 1 Change package in the technical evaluation section. Design changes are approved by the Director of Design Engineering or designee.

Individual tasks that are performed to accomplish design changes are also contained in the DCMs. These tasks are described below.

Calculations and analyses are controlled by procedures DCM-2 "Preparation and Control of Manual Calculations and Analyses" and DCM-14 " Preparation and Control of Computer Calculations." These procedures provide a format and process to be followed when completing a calculation or analysis. Integral to the performance of calculations and analyses is the requirement to describe the problem/objective/method, identify the design bases/ assumptions used, provide a summary/conclusion of the results, identify the distribution of the calculation, and identify components/related documents/related drawings affected by the calculation so that they can be identified for cross referencing in the Document Control System indexes. The methodology to be used in the performance of repetitive calculations is provided in the Engineering Standards Manual. This manual contains procedures that provide suitable methods for specific types of engineering disciplines such as calculating setpoints and instrument loop accuracies, pressure drop calculations, breaker and fuse coordination calculations, evaluation of combustible loading, and evaluation of local stresses for piping attachments. Calculations and analyses are checked by another individual other than the preparer and have an independent design verification performed using procedure DCM-4 "Design

Verification" on those calculations and analyses for QA Category I/M (safety related/augmented quality) SSCs. Calculations are approved by the cognizant engineering supervisor or designee.

Technical Procurement Specifications are controlled by procedure DCM-3 "Preparation and Control of Technical Procurement Specifications." This procedure is used for engineered items and services. Items that can be ordered from a manufacturer's Bill of Material or vendor catalog are excluded. Input for the specification is derived from design documents, applicable codes, standards, regulatory requirements, licensing commitments, QA/QC standards and acceptance criteria. Information to be provided in the specification is listed in the procedure and includes information such as references, technical requirements (materials, analyses, design and service requirements, electrical requirements, seismic requirements, tolerances), special processes, inspections/tests/examinations with acceptance criteria, cleanliness requirements, QA/QC requirements, and documentation required to be provided. Technical procurement specifications are reviewed using procedure DCM-12 " Review and Approval of NYPA Generated Technical Documents" unless the specification is for QA Category I/M item or service which requires an independent design verification using DCM-4 and QA concurrence. Specifications are approved by the Director of Design Engineering.

Identification and control of design interfaces are controlled by procedure DCM-6 "Design interface Control." This procedure is used to identify and control the design coordination for multi-discipline or multiple organization design engineering activities. Transmission of approved data is made by the controlled distribution of approved design documents or by the use of an Interface Control Document (ICD). ICDs are prepared by a Design Engineer (DE), reviewed by the Lead Design Engineer (LDE) for the project, and approved by the LDE's supervisor.

Reports and studies are prepared in accordance with DCM-7 "Preparation of Technical Studies and Reports." Reports and studies are prepared using the prescribed format which specifies those areas to be addressed. Generally, reports are prepared as a design report or a study report. Design reports describe the design, its intended function or method of operation. Study reports define alternatives and present the basis for alternatives and comparisons. Reports are typically formatted to include title, table of contents, introduction, main text/discussion, conclusion/evaluation, and recommendations. Reports are reviewed and approved using procedure DCM-12 " Review and Approval of NYPA Generated Technical Documents." Reports that are for QA Category I/M SSCs are design verified in accordance with DCM-4.

To prevent unnecessary delays, sometimes uncertain or incomplete design information is contained in design documents. DCM-10 "Tracking and Resolution of Holds" is used to identify and track this information. Design changes that make use of documents containing incomplete or uncertain information have the documents identified in a "hold log." Hold records are generated for the design document. The hold record identifies the design document that has the hold placed on it. In addition to the hold record, the design document is marked on the page with the incomplete or uncertain information, circling the information in question and writing in the hold record number. When the missing or incomplete information is updated into the design document, the hold is released and the log and design document is updated to reflect this change. The hold log is maintained by the LDE for the design change.

Design drawings are prepared in accordance with DCM-21 "Preparation, Review, Approval, and Control of NYPA Drawings." Design drawings may be either new drawings developed for a specific design change or an Interim Issue Drawing. Interim Issue Drawings are reproducibles made from plant original drawings with the design change depicted and highlighted along with original design information. Design drawings are prepared by the Design Engineering Design Section in accordance with the Design Drafting Standards (DDS). The LDE coordinates the input with the other design disciplines and provides the designer with the required information to complete the design drawing. Design drawings are checked by another designer for correctness. The LDE identifies the required reviews and the design supervisor issues the drawing for review. When the reviews are completed, the drawing is submitted to the Design Engineering Discipline Supervisor for approval.

Design verifications are performed for individual design tasks as required by the DCM procedures and also as an overall design change verification. Design verifications are performed using procedure DCM-4 "Design Verifications." DCM-4 provides guidance for performing design verifications by using either a design review method, alternate calculation method, or qualification testing method. The process provides for identification and documentation of multiple discipline verifications. Individual design document verifications are documented on the control sheets for the task and overall design change verifications are documented on the modification package or Type 1 Change cover sheets. The Design Engineering Discipline Manager/Supervisor is responsible for the design verification.

If required, field changes (Engineering Change Notices) ECNs are made to modifications and Type 1 Changes using procedure MCM-9 "Engineering Change Notice." This permits changes to be made in the field to design documents, installation specifications and procedures, and other modification documents. ECNs are not permitted to be used for problems identified outside the scope of the modification/Type 1 Change, to change the scope of the modification/Type 1 Change, to change the Nuclear Safety Screen or Evaluation, or change documents to a modification/Type 1 Change after it is closed out. ECNs are reviewed by the Responsible Engineer (RE) for the modification/Type 1 Change to ensure that they do not change the Screen or evaluation. ECNs are reviewed by affected on-site departments, QA for QA Category I/M, and a representative from the original design organization or Design Engineering discipline engineer if the original design organization is not available. ECNs are approved by the Design Engineering Discipline Supervisor. The RE is responsible for checking that design reviews and verifications are completed prior to completing turnover to the operations department.

PROCUREMENT DOCUMENT CONTROL

Procurement of materials, equipment, and services are controlled for new and replacement items used for modifications and maintenance to the nuclear facility. The Procurement Engineering Group is responsible for developing procurement documents and evaluating the acceptability of replacement items in accordance with procedure SED-AD-24 "Technical Evaluation of Components and Replacement Items." Items used in modifications have the requirements established in accordance with DCM-3 "Preparation, Review, and Approval of Technical Specifications" which are prepared by the Design Engineering Group.

Procured items (either new or replacement items) have the following attributes identified:

- Quality Assurance Category (safety related QA Category I , augmented quality QA Category M, or non-safety related Non-QA Category I. QA Category is determined using

procedures MCM-6 and SED-AD-24 or PEDB
- Technical Requirements (material, functional requirements, special processes, clearances, tolerances, etc.)
- Testing and inspections required.
- Documentation Requirements

Replacement items are divided into two categories, like-for-like and alternate items. Like-for-like items are those items that are physically identical in form, fit , function, and material composition of the original item. Alternate items are replacement items that are not identical to the original item. Alternate items must meet equivalency criteria specified in procedure MCM-14 "Type 1 Changes." If an equivalent item cannot be obtained, a modification must be performed and the specifications for the item developed in accordance with DCM-3. In identifying the technical requirements for items, this information is developed using SED-AD-24 based on checking design documents and data bases such as the FSAR, Plant Design Specifications, Drawings, Plant Equipment Data Base (PEDB), Environmental Qualification (EQ) List, and the Authority's Reg. Guide 1.97 response.

QA Category I Items are purchased from vendors who meet one of the following criteria:

(1) Already have an established Quality Assurance (QA) program meeting the requirements of Appendix B to 10CFR50 that has been audited by the Authority QA and placed on the approved vendor list.

(2) No Appendix B QA program established. Item will have Commercial Grade Dedication process performed.

Commercial Grade Dedications are performed using procedures SED-AD-24 "Technical Evaluation of Components and Replacement Items." The commercial grade dedication process is performed by identifying the safety function(s) and determining those critical characteristics that have measurable attributes that can be checked to provide reasonable assurance that the item will perform its safety function(s). A dedication method is established using special tests and inspections, commercial grade survey of the supplier, source verification, or acceptable supplier/item performance record or a combination of any of the methods. Upon satisfactory completion of the dedication method, appropriate documentation is generated and maintained for the component. The component is then installed in the nuclear facility.

Purchase orders for equipment or services that are categorized as QA Category I or M are required to be reviewed by the Quality Assurance Department prior to ordering the item.

INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Procedure Control

Procedures are used to govern the operation, testing maintenance, modification, design, and administration of the nuclear facility. Activities that require written procedures are listed in NRC Regulatory Guide 1.33, November 1972, Appendix "A"as required by section 6.8 of the IP3 Technical Specifications. Guidance for procedure controls are promulgated in the Nuclear Administrative Policies NuAP 6.2, "Procedure Hierarchy" and NuAP 6.3, "Procedure Use and Adherence." The requirements for procedure review and approval are contained in the IP3 Technical Specifications section 6.5.0. Procedure CMM 1.3 "Requirements for Preparation and Control of Nuclear Engineering Procedures" (corporate), and AP-3, "IP3 Procedure Preparation, Review and Approval," govern the process for controlling the preparation, review approval, issuance and revision of procedures. The "Procedure Writing Manual" (PWM) describes the standard formats for site procedures and supplements AP-3. AP-3 has undergone numerous revisions in the recent past as the result of both internal and external assessments of the procedure process.

Procedure changes may be accomplished as temporary referred to as (Term Procedure Changes) or permanent changes. The review requirements for both permanent and temporary requirements are the same. The review requirements require two qualified technical reviewers (QTRs) knowledgeable in the affected functional areas and additional cross-disciplinary reviewers as identified by the QTRs, and AP-3 cross disciplinary review determination. Additionally, if the procedure is identified as TSR (meaning it is required by Tech Spec 6.8), a 10 CFR 50.59 Safety Impact Screen is completed per MCM-4. Approval of permanent changes are made by the Procedure Sponsor in accordance with Technical Specification 6.5.0. Approval of temporary changes is made by two members of the plant staff, one of which holds an SRO license.

The Procedure Writing Manual (PWM) provides guidance on specific formats to be used for different procedures. Additionally it provides guidance on writing techniques, contents, and word usage and choice. For procedures that are used that require decision point or evaluations, guidance is provided on how to write acceptance criteria.

DRAWINGS

Design drawings are controlled as described under Design Control. Drawing updates are controlled as described under Document Control.

DOCUMENT CONTROL

Documents

NYPA generated documents are tracked within the Document Control system using a unique computer generated document identification number. This number is based on the specific document "type" (ie. Calculation, Analysis, Specification) in accordance with the applicable Design Control Procedure (DCM) governing the document type, and is tracked until the document is turned over to the Document Control group for file maintenance and distribution.

Documents received are checked for required approvals and are indexed into the Document Control system. Based on the document type and written request, standard distribution lists are linked to each document for distribution within the document control system. Distribution is controlled via a controlled document transmittal. Transmittals are required to be tracked for receipt acknowledgment and delinquency notices are generated for overdue acknowledgments. Periodic assessments are performed by the Document Control group to ensure that recipients of documents are properly maintaining and updating controlled files.

Document preparation and update processes for calculations, specifications, analyses, reports and studies are governed by the applicable DCM procedure. The preparer of the document or document revision is responsible for completing document verification (if applicable), checking that required reviews are accomplished and that approvals are obtained. New or revised documents are forwarded to the Document Control group for processing in accordance with CMM 4.3.

Vendor generated documents are accepted for use by the Authority in accordance with procedure DCM-11. This procedure contains the administrative controls for document tracking and provides guidance on those technical items to be checked to determine the acceptability of vendor documents. Vendor documents are required to be transmitted to document control for initial receipt and entry into the "DCM-11 Data Base" so that they can be tracked until they are accepted by the Authority. The responsible engineer for the task establishes the required reviews to be completed and coordinates the resolution of comments. Guidance for reviewers on information to be checked is listed on attachments to the procedure. Items to be checked include the following items such as: design bases, design input parameters, conclusions and design output, seismic, EQ, specifications, tolerances, incorporation of vendor equipment and design verifications. If the document is unacceptable, it is returned to the vendor to correct the identified problems. If the document is acceptable for use, it is accepted by the responsible engineer for the task and forwarded to document control. Document control enters the document's number in the Document Index Data Base and performs the required controlled distributions in accordance with CMM 4.3.

DRAWINGS

Drawing updates are processed in accordance with DCM-22 "Drawing Update Procedure." The Drawing Update program updates the plant drawings to incorporate changes made due to plant modifications or document discrepancies as identified via a Document Change Resolution (DCR). The program includes administrative controls for superseding and voiding drawings, as well as the process for entering new drawings into the plant Drawing Control System index. Guidance is provided for the scheduling of drawing updates using a drawing hierarchy scheme that prioritizes drawing updates based on its relationship to the safe operation and maintenance of the nuclear facility.

Upon receipt of the modification drawing package or DCR, the design group verifies the contents of the package against the drawing data base to check that all drawings are identified and all applicable outstanding changes have been posted or included in the package. The Drafting Supervisor schedules the updates based on the applicable drawing category. The drawing updates are completed by a draftsperson using the guidance contained in the Design Drafting Standards (DDS). Changes from the previous revision are highlighted by circling the changed information. The drawing updates are checked by a second individual for correctness, reviewed by the Design Supervisor, and approved by a Design Engineering Discipline Supervisor or designee.

VENDOR EQUIPMENT TECHNICAL INFORMATION PROGRAM (VETIP)

The VETIP program, addressing nuclear steam supply system (NSSS) vendors, was established as a result of NRC Generic Letter 83-28 and the Authority's commitments. The program has been modified to satisfy the requirements contained in NRC Generic letter 90-03 which expanded this program to include non-NSSS vendors. AP-18.1, "Controls of Vendor Equipment Technical Information" is the site procedure which defines and controls the overall process for VETIP. This procedure controls information receipt, evaluation for applicability to equipment, acceptance and incorporation in to plant processes, procedures and programs. Periodic vendor interfaces are controlled by individual departmental directives.

INFORMATION DATA BASES

Plant Equipment Data Base

The Plant Equipment Database (PEDB) is used to control and display source document information related to installed equipment. The PEDB Group is tasked with maintenance of the database which is accomplished using procedures AP-42 "Plant Equipment Data Base Program" and CIM-AD-5.2.12. Additions or revisions to the database are forwarded from various departments. The validation of PEDB information process includes an independent review utilizing approved document input sources. Feedback regarding plant changes comes from engineering and maintenance activities.

Recent additions to the PEDB consist of the following: the Environmentally Qualified equipment list and Maintenance Rule equipment. The Master Fuse List, Setpoint Control information, and Security System equipment are currently being converted to the PEDB.

Identified deficiencies in the PEDB have resulted in administrative processes being put in place to compensate for missing or incorrect information. These processes require component QA Category identification for those components missing this information. If this information is not identified, the components must be treated as safety-related (QA Category I) as per AP-9. Additional requirements include the field verification of component part numbers prior to working on plant equipment as per ICD-DD-01.

Electrical Cable and Raceway Information System

The Electrical Cable and Raceway Information System (ECRIS) is a computerized cable and raceway data management system for storing and retrieving electrical cable and raceway information. It provides information to support the engineering, design and modification of the plant Cable and Raceway System. It is a component of The Authority's computerized Integrated Nuclear Data Management System (INDMS). Design Control Manual Procedure DCM-25B, "Electrical Cable and Raceway Information System" covers the process and identifies the responsible organizations for maintaining the data base.

ECRIS stores and retrieves data according to the user's definable sort criteria or standard report forms via hardcopy or screen view. Typical information stored includes the following:

- cable and raceway identification, routing and parameters
- safety and system classification
- raceway fill and cable weights, designed and as-built
- continuity and function verification
- fire area identification

Updates and changes to the Cable and Raceway System ECRIS that change physical data are processed only through an approved modification, Type 1 Change, or Document Change Resolution (DCR) in accordance with AP18.8. Supplementary or editorial changes may be made for clarification purposes.

Fuse Lists

Fuse Control Procedure DEE-SD-01 is the directive that outlines the Fuse Control Program to ensure that the proper fuse is selected and installed in electrical circuits. The selection of the correct fuse size for a particular application is sized in accordance with engineering standards EES-3, EES-6 and EES-8 as applicable or an equivalent engineering evaluation. To ensure that only the correct fuse is installed in a circuit , DEE-SD-01 requires fuse verification for all fuses removed from permanent plant equipment during the performance of any work activity. Verification consists of ensuring that the installed fuse agrees with the Master Fuse List. In cases where a discrepancy exists between the installed fuse and the fuse listed in the MFL or is not identified on the MFL, DEE-SD-01 provides procedural guidance for the fuse discrepancy resolution. Fuse substitutions may be authorized by Design Engineering - Electrical. This authorization is not intended to bypass the modification process. Fuse replacements that affect the FSAR, Technical Specifications, or design basis shall only be implemented using an approved modification.

Changes to the MFL are required to be approved by a Design Engineering Electrical Supervisor or designee. The MFL is updated to maintain the data as current as possible to reflect the required configuration.

The MFL database will be transferred into the Plant Equipment Database (PEDB). This will allow plant personnel to have fuse data available "on-line" through the IP3 computer network.

CONFIGURATION INFORMATION DATA BASE

Controlled nuclear documents are indexed in two applications. The IP3 drawings application and the IP3 documents database. When a new revision is received, the old revision record is updated and a copy of the old revision record is automatically filed in the history database.

The software for the document control system was installed as a turnkey system to support the indexing and tracking of nuclear-related documents in 1984. The first database to be developed was the Records Management Data Base. The drawing database for IP3 was established in 1985.

In 1989, a controlled documents application was created. With the establishment of a Document Control group within the Nuclear Generation Department, a new controlled documents application was developed in June 1991. The new document application consists of a current revision database and a historical revision (or history) database. The new application includes the following features:

- A subsystem for the assignment of controlled document numbers
- Reference fields.(links to system, component, and other design documents)
- Film location field
- Fields for controlled distribution
- Electrical Change Control Form (ECCF) number
- Review, tracking, and distribution subsystem for vendor documents

MATERIAL DEFICIENCIES AND CHANGES TO THE NUCLEAR FACILITY

Problem Identification Description (PIDs) are used to document material deficiencies and are processed in accordance with SPO-SD-01 "Work Control Process." Deviation Event Reports (DERs) are used to identify non-conformance (such as procedural, design, Quality Assurance Program deviations, adverse trends, and surveillance test failures) and NSSS, A/E, and important industry events. DERs are processed in accordance with AP-8 "Deviation and Event Reporting and Operability Determinations." PIDs and DERs are reviewed to determine the effect on system operability. The DER and PID processes are discussed in the response to information requested in (d).

Changes to the nuclear facility may be required to improve plant safety or personnel safety and health, or as a result of a regulatory issue, or as a solution to a design deficiency identified by the NSSS supplier, A/E design firm, or Authority personnel. Changes also occur to correct plant material deficiencies for obsolete components and may be processed to improve plant availability or for economic reasons. Material deficiencies and, if required, changes to the facilities are processed under the following categories:

Corrective Maintenance
Document Change Resolution
Type 1 Changes
Modifications
Setpoint Changes
Temporary Modifications
Procedure Changes (discussed in the response to information request b).

CORRECTIVE MAINTENANCE

Those material deficiencies identified by a PID that can be resolved by corrective maintenance are processed by the Work Control and Maintenance Groups. These items are processed and resolved in accordance with SPO-SD-01 "Work Control Process."

Individuals discovering a plant deficiency, document the item using a PID and determine if there is an immediate operability or reportability concern. If operability or reportability is a concern the Shift Manager (SM) is notified immediately. Otherwise, once per shift, the SM reviews the PIDs generated for operability, reportability, and priority.

Normally, Monday through Friday, the PID Review Committee chaired by the Work Control Center (WCC) Supervisor, reviews new PIDs for operability, reportability, priority, and plant conditions required to complete corrective actions. The WCC then completes the PID and assigns a priority in accordance with AP-9 "Work Control." The PID is assigned to a responsible planning group or to the FIX-It-Now (FIN) team.

When the FIN team is assigned to complete the corrective action, the FIN team SRO performs a risk assessment and discusses the required plant configuration with the on watch Shift Manager, Control Room Supervisor, or Field Support Supervisor (FSS). The FIN team SRO determines the protective tagging requirements in accordance with the guidelines of AP-10.1 "Protective Tagging" and that the task is bounded by the current week risk assessment per SPO-SD-03 "On-Line Work Scheduling Process." The item is then isolated (if required), repairs made as required, and documented including any retests in the monthly minor maintenance package.

PIDS assigned to responsible planning groups have work packages generated for their accomplishment. When the packages are ready to work, a station work week schedule is prepared in accordance with SPO-SD-03. The WCC generates any required Protective Tagging Order (PTO) using procedure AP-10.1 and documents an operational review on the Operational Impact Sheet. The WCC SRO processes the work package in accordance with the approved schedule.

The PTO procedure controls the configuration of SSCs that are removed for scheduled maintenance or other operational tasks. This change in configuration is double verified at the time of application to check its correctness for systems identified in OD-35 "Independent Verification." After a PTO is generated, the WCC also identifies Technical Specifications, Operational Specifications, and Limiting Conditions of Operations (LCOs) to the Shift Manager. Also identified are expected plant conditions, components affected, and special instructions or sequencing that are required to execute the PTO. These are documented on an Operational

Impact Sheet. This procedure is also used to restore affected SSCs to their design configuration. Restoration of equipment is double verified to assure equipment is restored correctly for systems identified in OD-35 "Independent Verification."

REQUEST FOR DOCUMENT CHANGE (RDC)

Request for Document Change (RDC) is used to identify discrepancies between the as-built condition of the plant and the plant's design record and are processed in accordance with AP-18.8 "Resolving Apparent Document Discrepancies." RDCs are reviewed to determine if the discrepancy between the as-built condition of the plant is the result of the design record being improperly updated, or missed. If the discrepancy is the result of a failure to properly update the design document, the RDC is processed and a Document Change Resolution (DCR) is generated to reflect the as-built condition of the plant and the appropriate documents are updated. If the as-built condition of the plant is not the result of a failure to properly update the design document, the RDC is reviewed by a Technical Reviewer to determine if the technical aspects of the as-built condition support the design basis. Where necessary, calculations, analyses, studies, reports, design verifications and 10CFR50.59 screening are performed to confirm that the as-built condition of the plant supports the design bases. If the as-built condition does not support the design bases, a DER is processed to identify the design non-conformance. RDCs are approved by the Qualified Technical Reviewer.

TYPE 1 CHANGE

If a material deficiency requires a change to the facility because the component is obsolete, the deficiency may be resolved using the process in procedure MCM-14 "Type 1 Change" if the change meets the requirements of the procedure applicability determination. A Type 1 is defined as a change to a plant structure, system, or component (SSC) that does not change the overall design function, operation or critical characteristics of the SSC. This procedure is limited to changes that do not affect the design bases of the facility and by definition, does not require a 10CFR50.59 evaluation or affect Technical Specifications. The types of changes performed using this procedure are limited to Equivalent Changes or Type 1 Design Changes (small design changes). Small design changes consist of changes such as small changes to the routing of tubing and cabling, changes to mounting or support of plant equipment, or fuse rating changes usually necessary when equivalent replacement items are substituted.

MCM-14 controls change activities including preparation, technical evaluation, approval, installation, testing, turnover of the system to operations, update of required documents and data bases, and closeout. If the Type 1 change is a small design change, the design control processes contained in the Design Control Manual (DCM) and described in this section under the topic Appendix B to 10 CFR 50 are used to perform the necessary calculations, analyses, studies, and reports for the technical evaluation. Type 1 changes are reviewed by a designated engineer from the Design Engineering Department and other disciplines as determined by the Responsible Engineer. Design Changes are verified in accordance with DCM-4 "Design Verification" if the change is for safety related (QA Category I) or augmented quality (QA Category M) systems, structures and components and are approved by the Director Design Engineering or designee.

MODIFICATIONS

In determining if a modification to the facility is required or warranted, the policy is to minimize plant modifications while assuring the plant operates as designed through maintaining an effective preventative and corrective maintenance program. A review committee considers recommendations to modify the facility, considering the attributes of safety, commitments to outside agencies, personnel health and safety, and plant availability and economics. Recommended modifications are prioritized and maintained on a list. Procedure ADM-SD-16 "Engineering Work Ranking System" is used to aid in making this determination.

Approval to commence work for modification package development is in accordance with MCM-17 "Initiation of Modifications." Approval is recommended by the Manager System Engineering, Director Design Engineering, and final approval granted by the Plant Managers.

Modifications and Minor Modifications are performed in accordance with procedures MCM-3 "Modification Package Preparation, Review and Approval" and MCM-5 "Minor Modifications." A Preliminary Engineering Package (PEP) is developed in accordance with MCM-2 "Preliminary Engineering Package." The PEP is used to obtain preliminary concurrence from different organizations and establish preliminary design bases and Quality Assurance/Quality Control Design Requirements. The PEP may be waived at the discretion of the Director Design Engineering or the VPE-PC.

The Minor Modification procedure (MCM-5) is used for modifications that; (1) are limited in scope requiring minimal engineering effort, (2) require small amount of resources, and (3) have readily defined and a limited number of physical interfaces with other plant structures,systems, and components. Both MCM-3 and 5 address the same elements for the performance of a modification and are addressed below.

<u>Elements of modification preparation (MCM-3 and 5) consist of the following:</u>

(1) Establishment of the division of responsibilities to perform the various activities associated with the modification. For modifications performed using MCM-3 this includes the use of a Modification Responsibilities List (MRL).

(2) Development of the modification scope, description, and identification of the systems and components.

(3) Detailed design preparation in accordance with DCM procedures. (Description of Design Control is described under Appendix B to 10CFR50)

a. DCM-2, calculations and analysis
b. DCM-3, technical specifications
c. DCM-4, design verifications
d. DCM-6, design interface control
e. DCM-7, studies and reports
f. DCM-10, tracking and resolution of design holds
g. DCM-11, reviewing and accepting vendor documents
h. DCM-12, reviewing in house documents
i. DCM-13, conduct of engineering
j. DCM-14, computer calculations

k. DCM-21, preparing design drawings.

(4) Completion of the 10CFR50.59 process using MCM-4.

(5) Notification to other departments of the modification providing pertinent information to facilitate changing other documents as required. Identification of documents and procedures that require changing.

(6) Input recommendations from operating and maintenance organization for construction, operability, and maintainability.

(7) Walkdowns of the physical areas affected by the modification.

(8) Development of installation requirements by the design organization in accordance with MCM-10 "Preparation of Engineering Requirements for the Installation of Modifications."

(9) Development of modification test requirements by the design organization in accordance with MCM-11 "Preparation of Modification Test Requirements."

(10) If required, setpoint changes are provided in accordance with MCM-8 "Setpoint Control."

(11) If required, software changes are accomplished using the guidance in the CMM 5.1 series procedures for Software Quality Assurance. Because of the unique design considerations for software, additional requirements are established including a software control plan, verification, and validation.

Modification packages that are prepared by external organizations are accepted by the Responsible Engineer in accordance with MCM-12 "Review and Acceptance of Modification Packages." Documents not part of a modification package are accepted in accordance with DCM-11 "Control, Review, Comment and Acceptance of Vendor Documents." The DCM-11 process may also be used for individual documents of a modification package that are received as a complete modification package.

Modification packages are reviewed by departments indicated for review by the Responsible Engineer or Lead Design Engineer. Design Verification of Safety Related (QA Category I) and Augmented Quality (QA Category M) Modification packages are conducted in accordance with DCM-4 "Design Verification." The Quality Assurance Department reviews as a minimum all QA Category I and M modification packages. Modification packages are presented to the Plant Operations Review Committee (PORC) for concurrence and are approved by the Site Executive Office or designee.

Modifications are installed in the facility in accordance with work control procedure SPO-SD-01. Those requirements for installation developed during the preparation of the modification package are translated into installation instruction in accordance with departmental directives. The Responsible Engineer for the modification ensures that installation requirements are covered in the installation instructions.

Test procedures are prepared in accordance with AP-3 "IP3 Procedure Preparation, Review, and

Approval." Existing procedures (e.g. operating, maintenance and surveillances) may be used to test the modification or parts of the modification if the procedure checks required testing requirements. The Responsible Engineer for the modification ensures that test requirements are covered in the test procedures.

Modified systems are returned to service using procedure MCM-19 "Modification Turnover and Closeout." This procedure controls those tasks required to be completed prior to turning the system over to the Operations Department. A checklist is used to track turnover items for installation, operations, training, and engineering. The checklist identifies those items necessary to be completed prior to the Operations Department declaring the system operable. Items from this list that are necessary for a system to be declared operable that have not been completed at turnover to operations are tracked using the Work Request system. The Responsible Engineer for the modification is responsible for ensuring that this task is completed and system acceptance is approved by the Operations Manager or designee.

Modification closeout is performed using procedure MCM-19. Modification closeout is used to establish tracking of those documents and data bases that have been identified for updating. A checklist is used to track the submittal of information to those departments responsible for the updates. Those items that are required to be completed promptly are identified for completion prior to modification closeout. Those items that are less critical, are tracked using the Action Commitment Tracking System (the ACTS data base is a system to track items that need to be completed) and are updated after the modification is closed out. The Responsible Engineer for the modification is responsible for ensuring the documents and data bases requiring update are completed by the responsible departments or are tracked in the ACTS system. Modification closeout is approved by the Design Engineering Supervisor.

SETPOINT CHANGES

Setpoint Changes are processed in accordance with MCM-8 "Setpoint Control Program." If a hardware replacement is also involved, the Setpoint Change is processed in addition to the modification or Type 1 Change. When no hardware replacement is required, MCM-8 is used by itself to control preparation, technical evaluation, approval, installation, testing, turnover of the system to operations, update of required documents and data bases, and closeout. 10CFR50.59 considerations are checked by using procedure MCM-4. Technical evaluations for setpoint changes are performed in accordance with Engineering Standard procedure IES-3 "Instrument Loop Accuracy and Setpoint Calculations" and other appropriate standards. Setpoint changes are reviewed by System Engineers, and Quality Assurance (QA I and M). The level of approval required is based on the Setpoint Type and QA Category. Setpoint Type 1 and 2 (Reactor Protection System, Engineered Safeguards Actuation System, required by Technical Specifications, Regulatory Guide 1.97 and Appendix R) and QA Category I are reviewed and approval recommendation made by the Plant Operating Review Committee (PORC) and approved by the Plant Manager. All other Setpoint Changes are approved by the Director of Design Engineering.

TEMPORARY MODIFICATIONS

A temporary modification is generated to document temporary physical or functional changes to plant SSCs which are not described in approved plant documents. Temporary modifications are performed in accordance with AP-13 "Temporary Modifications." AP-13 requires that temporary modifications be reviewed on a periodic basis (not to exceed three months) to ensure that they are actively being pursued to be cleared. Quarterly, the System Engineering Manager is required to submit a report that evaluates temporary modifications to check that they do not pose an aggregate safety problem. 10CFR50.59 considerations are checked by using procedure MCM-4. Configuration of the plant is maintained by updating the "Type A" drawings in accordance with SED-AD-1 to reflect the temporary modification and the changing of operating procedures to accommodate any new instruction required.

10 CFR 50.59 IMPLEMENTATION

The requirements of 10 CFR 50.59 are implemented via a Modification Control Manual (MCM) procedure, MCM-4 ("Nuclear Safety and Environmental Impact Screens and Nuclear Safety Evaluations"). MCM-4 is used in conjunction with other procedures which control a variety of activities such as modifications, procedure changes (operating, maintenance, surveillance, administrative, design change, etc.), and temporary modifications. MCM-4 uses the industry guidance provided in NSAC-125.

MCM-4 uses a two part process. The first part is for screening an activity and, the second part if required, is for performing a detailed safety evaluation. The screening process is used to determine if an activity needs to have a Nuclear Safety Evaluation (NSE) performed. Some activities have applicability determinations built in to the process and do not require a screen because the activity is limited and do not have a 10 CFR 50.59 impact. An NSE is used to determine if the activity involves an Unreviewed Safety Question (USQ).

Those individuals who perform MCM-4 tasks (preparers, reviewers, and approvers) are required to be qualified through the site's training programs for 10CFR50.59.

SCREENS

Nuclear Safety and Environmental Impact Screens (NSEIS) are used to determine if a proposed activity requires a 10CFR50.59 Nuclear Safety Evaluation, a Technical Specification Amendment, or an Environmental Impact review. NSEIS consist of five questions which are broken into multiple parts and the answers for which are prepared by a qualified designated individual following the requirements and guidance of the procedure.

The first three questions pertain to the Safety Analysis Report (SAR) which is defined as the FSAR and any approved NSEs that change the information presented in the FSAR. The FSAR sections that were reviewed in answering the NSEIS questions are required to be listed. The first part of each of these three questions determines if the activity being performed is; (1) described in the SAR, or (2) affects SSCs described in the SAR, or (3) involves a test. Written guidance is provided to the preparer to determine what needs to be considered when deciding if the information is described in the SAR. Items not specifically discussed in the SAR are evaluated to determine if they; (1) are part of a larger item that is described in the SAR or (2) affect the design and function of an item described in the SAR. A "YES" answer to the first part of any of these three questions requires that the remaining parts of the question be answered. The remaining parts determine if the activity being performed is consistent with the requirements of the SAR. Written guidance is provided to the preparer in answering these questions. The questions are posed such that a "YES" answer requires a NSE to be performed. "NO" answers do not require an NSE but do require written justification as to why the activity is consistent with the requirements of the SAR.

The fourth question pertains to the facility Technical Specifications (TS). The first part of this question determines if the activity involves items described in the facility TS. The TS sections that were reviewed in answering the questions are required to be listed. A "YES" answer to this part of the question requires that the remaining parts be answered. The remaining parts of the question determine if the activity is consistent with the TS. "YES" answers require that a Technical Specification Amendment be processed in accordance with Licensing Procedures. "NO" answers require written justification as to why the activity is consistent with the requirements of the TS. Operational Specifications are reviewed for the same considerations. Any required changes to Operational Specifications are recommended by the Plant Operations Review Committee (PORC) and approved by the Plant Manager.

The fifth question determines if the task has radiological or a non-radiological environmental impact. A "YES" answer requires notification of the Radiological Environmental Services Manager for the preparation of an Environmental Impact review.

NSEIS are approved by qualified designated individuals. The approver checks for completeness, procedure compliance, and sound logic of the written justifications and conclusions.

NUCLEAR SAFETY EVALUATIONS (NSEs)

NSEs are prepared for those activities that have been determined to possibly involve an Unreviewed Safety Question (USQ). An USQ exists if the proposed activity does any one of the following; (1) Increases the probability of the occurrence of an accident previously evaluated in the SAR, (2) Increases the consequences of an accident previously evaluated in the SAR, (3) Increases the probability of a malfunction of equipment important to safety previously evaluated in the SAR, (4) Increases the consequences of a malfunction of equipment important to safety previously evaluated in the SAR, (5) Creates the possibility of an accident of a different type than any previously evaluated in the SAR, (6) Creates the possibility of a malfunction of equipment important to safety of a different type than any previously evaluated in the SAR, or (7) Reduces the margin of safety as defined in the basis of any Technical Specification.

NSEs are prepared (1) if required by an NSEI screen, or (2) without performing the NSEI screen, if it is already known that the activity needs to be checked for an USQ. The process requires that a qualified designated individual prepare the NSE by researching TS, SAR, NRC Safety Evaluation Reports, docketed NRC correspondence, Design Bases Documents, Operations Specifications Licensing Commitment Data Base, and the Action Commitment Tracking System (ACTS). Using the written guidance provided, the preparer answers the seven questions and establishes if an USQ exists for the proposed activity. References, FSAR, TS, and other documents reviewed are required to be listed. Those FSAR sections that need revision based on the activity are listed and an FSAR change request is required to be prepared in accordance with Nuclear Licensing Procedure NLP-3. If an USQ has been determined for the activity, the activity must be either canceled or revised so that no USQ exists, or a request made to the Director Nuclear Licensing to obtain NRC review and approval prior to activity implementation.

An NSE is reviewed by other departments and disciplines if it involves more than one type of activity (e.g. Operations and Maintenance), or more than one design aspect (e.g. Mechanical, Electrical, Civil/Structural, Instrument and Control, and Fire Protection). A qualified designated individual reviews the NSE for sound logic, technical accuracy, and reasonable conclusions ensuring that the guidance provided was followed, cross-department/discipline reviews for the NSE are adequate, and that no USQ exists. Copies are distributed to the Plant Operating Review Committee (PORC) members for review, and their comments are resolved prior to presenting the NSE at the PORC meeting for concurrence. NSEs are approved by the Plant Manager or designee. A post implementation review of the NSE is conducted by the Safety Review Committee (SRC).

REASONABLE ASSURANCE OF SAFETY (RAS) AND JUSTIFICATION FOR CONTINUED OPERATION (JCO)

In those instances where information is missing precluding the completion of a NSE, a JCO or RAS is completed using procedure AP-25.5. This procedure was developed using the guidance provided in Generic Letter 91-18. A RAS or JCO is an interim evaluation to promptly address safety issues until all required information is available and a NSE can be completed. If a condition prohibited by a Technical Specification exists or will exist, a JCO is prepared, otherwise a RAS is prepared.

The JCO or RAS is prepared in the same format with the same cover sheet, and is reviewed and approved in the same manner as a NSE. The JCO or RAS is required to identify all outstanding open items that are needed to answer all of the NSE questions. Each of the open items is required to be tracked to completion using a Work Request (WR) or an Action Commitment Tracking System (ACTS) as appropriate. The JCO or RAS is assigned an expiration date. If the NSE cannot be completed by the expiration date, a revision is issued to the JCO or RAS with a new list of items remaining to be completed and a new expiration date is established. Updates are provided to PORC on the status of RAS/JCO approximately every 30 days.

The RAS and JCO address the potential effects on the plant's design bases as reflected in the FSAR and the licensing bases contained in the FSAR, Technical Specifications, Operational Specifications, docketed commitments in NRC correspondence, and Safety Evaluation Reports (SERs).

10CFR50.71(e)

In accordance with the guidance provided in Nuclear Administrative Policy (NuAP) 4.9, it is management's expectation that all Nuclear Generation personnel are responsible for the accuracy of the FSAR and for identifying discrepancies to Nuclear Licensing.

The process for updating the FSAR is controlled by procedure NLP-3 "FSAR Updates." This procedure requires that the FSAR is updated to reflect plant modifications, changes to procedures described in the FSAR, 10CFR50.59 Safety Evaluations, Technical Specification Amendments, NRC correspondence, and to reflect the on resolution of discrepancies.

The procedure requires the following actions be accomplished:

(1) Requests are sent to the Responsible Engineers for modifications, Nuclear Safety Evaluations, and Technical Specification Amendment submittals to review their documents to determine if the FSAR needs to be changed. These individuals are responsible for providing input in the form of FSAR markups with the changes.

(2) NRC correspondence is reviewed for FSAR impact and FSAR updates are generated as required.

(3) FSAR section experts are assigned for each update cycle. The FSAR markups submitted as a result of the above activities are compiled by section and sent to the experts for review. The review ensures that the markups accurately reflect the supporting change documents and that different markups affecting the same section do not conflict.

(4) At the completion of the reviews, the FSAR sections are finalized and the FSAR is updated to reflect the changes.

In addition to the requirements of NLP-3, procedures for other activities (Modification Closeout, Nuclear Safety Evaluations, Setpoint Changes, and Temporary Modifications) that could change the FSAR have statements in them that require individuals responsible for the other tasks to provide input for updating the FSAR.

TECHNICAL SPECIFICATION CHANGES

BACKGROUND

The Technical Specification (TS) change process is governed by 10CFR50.90, 50.91, and 50.92. The TS submittal and amendments are processed in accordance with NLP-2, AP-2, AP-18.7 and SRCP-8. A TS change request can be initiated by any employee in accordance with AP-18.7. This vehicle ensures that personnel from all disciplines have a method for initiating a TS change.

Once the TS request is approved, the Licensing Group processes the amendment request for NRC submittal. The submittal process includes steps to ensure that the TS changes do not involve a significant safety hazard. These steps are as follows.

- NLP-2 requires the preparer to confirm that all affected TS sections have been identified.
- Once the submittal is drafted, it is distributed for review to the Plant Operating Review Committee (PORC) and other cognizant personnel in accordance with AP-18.7. This review requires the addressees to identify the documents affected and the training or modifications required as a result of the amendment proposal. Actions identified as a result of this review are incorporated into the commitment list attached to the submittal. In accordance with NLP-2, the actions in this commitment list are entered into the plant's action tracking data base (ACTS) to track their completion.
- The draft amendment proposal is presented to PORC and the Safety Review Committee (SRC) for review and concurrence. In accordance with AP-2, PORC reviews the submittal to ensure nuclear safety issues are addressed, the submittal is technically accurate and understandable, and commitments are reasonable. In accordance with SRCP-8, the SRC reviews the TS amendment proposals to assure that they do not degrade the present safety design bases of the plant as stated in the FSAR, do not violate other plant TS, License Conditions, or Orders, do not affect the environmental impact of the plant, and that no unreviewed safety questions have been created as defined in 10CFR50.59.
- Once the amendment has been submitted and approved by the NRC, the approved amendment is sent to the affected department managers. This serves as a reminder that TS implementing and follow-up actions as associated with the amendment must be completed. Any additional items to be completed are to be entered into the Action Commitment Tracking System (ACTS).

(b) Rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures at Indian Point 3

The Authority is confident that adherence to the processes described in this document provide reasonable assurance that design bases requirements are being properly translated into operating, maintenance, and testing procedures and that, if inconsistencies are found, they are evaluated and proper corrective actions are taken. The Authority bases this confidence on the quality of these processes in addition to the various efforts, reviews, inspections, audits and walk-downs that have been performed to provide consistency between the design bases and plant procedures.

PROCEDURE PREPARATION REVIEW and APPROVAL

The primary mechanisms for translating design bases information into operating, maintenance, and testing procedures at Indian Point 3 are; (1) control of procedure development and revision to incorporate existing design bases information, and (2) proper identification of changes required to these procedures during design activities which create, clarify or modify the plant design bases.

The initial procedures at Indian Point 3 (IP3) were prepared in accordance with NRC Regulatory Guide 1.33, November 1972, Appendix "A" as required by section 6.8 of the IP3 Technical Specifications. AP-3, "IP3 Procedure Preparation, Review and Approval," governs the process for controlling the preparation, review, approval, issuance and revision of procedures. The "Procedure Writing Manual" (PWM) describes the standard formats for site procedures and supplements AP-3. The procedures and administrative controls have been improved as part of the Restart and Continuous Improvement Plan (RCIP) where substantial upgrades were made. Changes to these procedures provided improvements for performing technical and safety reviews for new and revised procedures.

Each program and procedure required by Technical Specification 6.8 and other procedures that affect nuclear safety, and changes thereto, receive a Technical and Safety Review. When appropriate, revised procedures also receive field walk through to verify their accuracy. Cross disciplinary reviews are conducted, if determined necessary by the Technical Reviewer. Prior to approval, procedure changes receive a safety screening review (10CFR50.59 applicability review) by a Qualified Safety Reviewer (QSR) in accordance with procedure MCM-4. If the QSR review determines a Nuclear Safety Evaluation (NSE) is required, then the NSE is prepared to verify that the change does not involve an Unreviewed Safety Question (USQ) or a Technical Specification Change. NSEs are submitted to the Plant Operating Review Committee (PORC) for review and recommendation for approval. This process of required multiple reviews and approval is to provide confidence that the procedure is valid and the design bases requirements are translated into operating, maintenance, and testing procedures to comply with the rules of 10CFR50.59 and Appendix B of 10CFR50.

Design bases changes or clarifications occur through a variety of methods including: Document Change Requests (DCRs), Type 1 Changes, Modifications, Setpoint Changes, Temporary Modifications, Calculations, and Design Document Open Items (DDOI) Evaluations. IP3's design control and configuration management procedures identify the actions necessary for translating changes to the plant design bases into operating, maintenance, and testing procedures as well as other plant documents and programs.

The design change process at IP3 is controlled through APs and MCMs as discussed in our response to information request (a). These procedures require design information to be transmitted from engineering to operations, maintenance, and testing procedure personnel for review. This design information is reviewed to determine impact on operating, maintenance and test procedures and associated programs. Changes to these procedures are controlled under station administrative procedures, which require the review, approval and issuance of affected procedures, as part of the modification turnover process to operations.

The Technical Specification change process at IP3 is controlled through APs and NLPs as discussed in our response to information request (a). Proposed Technical Specification amendments are forwarded to potentially affected organizations, and are reviewed for their effect on procedures. Implementation of required procedure changes are assigned ACTS items and tracked to support amendment approval.

Audits, surveillances, and assessments of the procedures are conducted by the responsible organizations, NYPA s Quality Assurance, contractors and the NRC. Deficiencies identified are required to be documented and resolved.

New efforts are regularly being initiated to improve the overall quality of the procedures and the programs at Indian Point 3. These efforts include, Procedure Upgrade / Validation Projects, Setpoint Verification Projects, and Reviews of Industry Operating Experience. Discussions of initiatives associated with Operating, Instrument and Control, Maintenance and Surveillance and Test Procedures are included in this response as follows:

OPERATING PROCEDURES

Procedure Reviews

<u>Pre-Start Up Walk-downs</u>

Prior to plant restart in 1996, Plant Operating Procedures (POPs*)*, Alarm Response Procedures (ARPs), Off Normal Operating Procedures (ONOPs) and System Operating Procedures (SOPs), needed for restart, were reviewed and upgraded. This effort included field configuration reviews by plant operators to verify procedure correctness and an additional review by a Senior Reactor Operator. Procedures were revised, as necessary, to reflect their input. During the process, various design and licensing documents were reviewed, as appropriate, including the FSAR and Technical Specifications. Management's expectation of strict compliance was reiterated to station personnel which added emphasis on the need for accurate usable procedures. Operator feedback forms and Temporary Procedure Changes (referred to as Term Procedure Changes at IP3) were incorporated which improved the overall technical accuracy and usability of the procedures.

Review of Safety Screens for Operating Procedures

In late 1995 and early 1996, the Independent Safety Engineering Group (ISEG) at NYPA conducted an extent of condition review for inadequate safety screens for Operating Procedure Revisions. The purpose of the review was to determine if the MCM-4 Safety Screens were performed during the time when administrative procedure, AP-3, allowed MCM-4 50.59 screening to be bypassed for non-intent changes. The review determined that there were weaknesses in the process and in a number of cases procedure changes were issued without undergoing a required 10CFR50.59 screening. To address the deficiencies, 10CFR50.59 screens were performed for deficiencies identified in procedure changes still in effect at the time of the review. As a long term corrective action, the process was amended to require 10CFR50.59 screening for procedure changes, except for editorial changes. The identified deficiencies were evaluated as satisfactory, justified through analysis, or corrected to ensure they were properly resolved.

Ultimate Heat Sink Analysis

As a result of this review, plant procedures were reviewed by Westinghouse and recommended changes were implemented to verify licensing bases requirements were properly translated. The changes made in these revisions included setpoints and valve positions for the affected Alarm Response Procedures and Check Off Lists. In July 1995, Headquarters Reactor Engineering conducted an independent audit of the procedural revisions at Indian Point 3 to support the Ultimate Heat Sink analysis and concluded that design bases as well as recommended requirements had been adequately translated and implemented.

PROCEDURE VERIFICATIONS

Review Of Accident Analysis Assumptions

In the last quarter of 1995, a review was performed by Headquarters Reactor Engineering which compared a selected scope of operating procedures against assumptions made in the Accident Analysis Basis Document (AABD). The review focused on readily measurable and controllable operating parameters that were initial conditions for design bases accidents. The review checked Surveillance Tests, Alarm Response Procedures, Plant Operating Procedures, System Operating Procedures and Off Normal Operating Procedures. None of the steps in these procedures caused the plant to be operated in a manner inconsistent with the accident analysis.

Training on the Core Operating Limits and assumptions used in the accident analysis of IP3 for Design Bases Events was given to Plant Operations Review Committee members, plant operators, and managers and supervisors of technical groups. Similar training is being given to Engineers as part of the Engineering Support Personnel (ESP) program.

Emergency Operating Procedures (EOP)

The Westinghouse Emergency Response Guidelines (ERG) Revision 1 were first issued in the early 1980s. The Authority developed EOPs based on Westinghouse ERGs. The operations staff worked in conjunction with Westinghouse to develop values for ERG setpoints required within the procedures.

During the mid to late 1980's, Equipment Qualification (EQ) in post accident environments was addressed and the EOP setpoints were reviewed against EQ criteria and revisions made as appropriate.

As part of 480 V Emergency Bus Electrical Load Studies, Engineering and Operations worked together to develop loading criteria when shifting from the Injection Phase to the Recirculation Phase for postulated design bases events. During this project, electrical calculations, load studies, and short circuit analyses were conducted with regard to vital bus load sequencing. These studies resulted in procedural changes.

In July 1996, the EOP Setpoint project was initiated to verify that setpoints used in Indian Point 3 EOPs are supported by an approved Design Calculation or a Westinghouse provided setpoint within the Westinghouse generic Emergency Response Guidelines. In addition, an EOP Setpoint Manual is being developed which defines the scope, format, approval process and configuration control for EOP setpoints. This manual will provide the necessary interface to the MCM-8 Setpoint Control process to maintain existing EOP setpoints and setpoint changes consistent with the design bases of the plant. The project is scheduled to be completed by mid 1997.

An initial review of the EOP Setpoints and the Setpoint Manual was performed in conjunction with the EOP revisions to support the 24 Month Cycle Extension. The EOP Setpoints were evaluated regarding instrument accuracy. This review identified two setpoints in the Critical Safety Function Status Trees that were incorrect in the controlled version of the EOPs. The deficiencies were corrected with the issuance of the EOP revision to support the 24 Month Cycle Extension. A review of the draft EOP Setpoint Manual Attachment 1 was completed in December, 1996. This review identified 28 setpoints which required additional supporting documentation to validate the setpoints or its basis. These discrepancies were reviewed to determine operational significance. Only 1 setpoint which dealt with RCS Pressure Temperature limits to prevent brittle fracture was determined to have potential operational impact and was documented and evaluated in a Deviation and Event Report (DER). The applicable EOP will be updated when the setpoint change is processed in accordance with MCM-8. Two additional DERs are tracking five setpoints with conflicting basis which are under review. The remaining 22 setpoints are scheduled to be further reviewed and resolved.

Plant Labeling Program

The Plant Labeling Program, Operations Directive-4 was developed based on NUREG-0700, EPRI NP-6209, and INPO 88-009 in 1986. The process requires reviews of equipment labeling against P&ID drawings and the Plant Equipment Database to determine the correct equipment identification for valves. Any changes are then reviewed against and translated into the affected operating procedures and drawings, as appropriate.

Operations Procedure Upgrades and Verifications

In August of 1996, the Operations Department initiated a validation and verification project to review procedures to verify that the procedures adequately address design and license bases requirements. This project is part of the long term improvement plan to complete an independent, organized and thorough review of operating procedures and to document the bases for stated actions.

The project deals with approximately 800 procedures which includes; Plant Operating Procedures (POPs), System Operating Procedures (SOPs), Alarm Response Procedures (ARPs), Off Normal Operating Procedures (ONOPs), Periodic Testing Surveillance Procedures (PTs), Operations Directives (ODs), Operator Log Sheets and Graphs. EOP's are not included since they are addressed in a different project.

This project is intended to provide traceable documentation to validate that Operations Department procedures operate and test the plant as required by the design and licensing bases, as well as identify any discrepancies that require resolution.

Procedures are grouped into three areas of focus. The initial focus will be on the systems detailed in report HS 1026-01, Probabilistic Risk Assessment Application at NYPA. This report addresses the hierarchy of plant systems contributing to safety and consists of a total of 23 systems. The second focus will be on the systems mentioned in the FSAR and Technical Specifications and consists of a total of 48 systems. The third focus will be on non-safety related systems and miscellaneous procedures that are not tied to any one system and consists of approximately 30 systems.

Currently a total of 20 procedures (15 for Auxiliary Feedwater, 2 for 125V DC, 2 for 480V AC, and 1 for Main Steam) have received a preliminary review as a pilot exercise. The review identified only minor discrepancies between the FSAR, Design Bases Documents (DBD) and procedures. Reviews to date have not identified any issues that would indicate that any structure, system, or component has been operated or tested outside its design bases. The project is scheduled for completion by mid 1999.

As a separate but related initiative, a review of procedures to provide consistency in human factoring, format, content, and wording is being conducted to achieve a uniform high standard of procedures and operator performance. This project is approximately 65% complete and is scheduled for completion by the end of 1998.

Training Simulator

Programming of the Indian Point 3 simulator includes the applicable design bases parameters presented in the plant design and licensing documents. The simulator is used to simulate plant response for training and examination of licensed operators and Shift Technical Advisors. Recognition, response and condition mitigation is performed using approved operating procedures, including SOPs, ONOPs, and EOPs. Use of these approved operating procedures for training assists in validating their technical accuracy and sequence of step presentation for in-plant use.

INSTRUMENT & CONTROL (I&C) PROCEDURES

PROCEDURE REVIEWS

I&C procedures are reviewed on a periodic basis in accordance with Administrative Procedure (AP-3). This review is to verify that the procedures address and implement the requirements made in the FSAR, Technical Specifications, Quality Assurance and NRC Regulations. New and revised procedures are required to undergo a 10CFR50.59 safety screen in accordance with MCM-4. This review verifies that the design, operation, and function of structures, systems, or components described in the FSAR will not be altered, nor will the changes cause structures, systems, or components to be operated, in a manner that would violate design and license bases requirements.

The scope of this I&C procedure program includes approximately 500 Non Technical Specification Calibration Procedures and 200 Technical Specification Surveillance Procedures. The schedule of priority of the revision process is based on the periodic re-review dates of the procedures, Temporary (Term) Procedure expiration dates, and plant scheduling requirements.

Approximately 70% of the I&C procedures have been completely reviewed. The remaining 30% of these procedures will be reviewed as periodic review dates come due. The Technical Specification portion of these remaining procedures is scheduled to be completed prior to restart from Refuel Outage #9 in 1997. The Authority is confident that adherence to the processes described in this document provide reasonable assurance that design bases requirements are being properly translated into I&C procedures and that, if inconsistencies are found, they are evaluated and proper corrective actions are taken.

PROCEDURE VERIFICATIONS

Technical Specification Cross reference Matrix

A Technical Specification Cross Reference Matrix was developed to directly relate specific Technical Specification Surveillance requirements to the applicable surveillance test procedures. The development of this matrix provided confidence that for each Technical Specification Surveillance Requirement there is an associated surveillance procedure.

Plant Instrument Calibration List, PFM-64

PFM-64 was developed to provide a formal listing of installed plant instrumentation which are required to be calibrated to support operability determinations associated with functional surveillance test procedures performed by Operations, Performance, I&C, and Fire and Safety Departments.

Surveillance Test Start-Up Pre-Requisite List, PFM-49

This list was developed to provide a mechanism for identifying the status of scheduled periodic Surveillance Tests to be performed prior to achieving plant milestone conditions during start-ups following outages. This list is provided to assist operators in determining operability of plant equipment following an extended outage.

Setpoint Verification

In response to identified deficiencies with setpoint control, the Authority instituted a Setpoint Control Program. This program consists of controls for setpoint changes, creation of a setpoint data base, methodology for reconstituting the bases for setpoints, the establishment of setpoint types, and the verification of selected setpoints. This program was originally started in 1993 and included the establishment of process controls for setpoints including change control, effect on plant operating procedures and updates, setpoint implementation in the facility, and document updates. Procedure MCM-8 "Setpoint Control" was written to control these functions.

The initial identification and cataloging of setpoints (and categories thereof) was completed but due to limited resources the compilation of setpoint and setpoint related information was not completed for all components in the identified categories. The Interim Setpoint Data Base (ISD) as it exists at this time includes setpoints for Type 1 functions, a substantial portion of Type 2 and some Type 3; all within the general category of instrumentation. Type 1 Setpoints include Technical Specification values for the Reactor Protection System and Engineered Safety Features; Type 2 includes FSAR and Technical Specification setpoints not identified as Type 1 that are required for operability of equipment required by Technical Specifications or Regulatory Guide 1.97 instruments and Appendix R Equipment: Type 3 includes system related setpoints not identified as Type 1 or 2. Within these groups, 616 setpoint datasets were compiled and the collection was identified within the document Interim Setpoint Data Base (ISD). Where available, approved calculations were listed with the setpoint data in the ISD. The bases for these calculations were not verified at the time that this database information was collected.

In late 1995, the Plant Manager created a Setpoint Control Committee in response to an NRC inspection that noted setpoint control as a problem. This committee established an action plan IDEE-APL-003 for improvement of setpoint control. This action plan was approved in early 1996 and included those tasks necessary to establish satisfactory control of setpoints. These tasks included:

- Verifying setpoints in the ISD (verification included reviewing the setpoint for correctness with current established conditions in the FSAR, Tech Specs, Setpoint Change Requests (SCRs), calculations, calibration procedures and surveillance tests).

- Transferring setpoints from the ISD to the Plant Equipment Data Base (PEDB).

- A consistency review of setpoint data between Upper Tier Documents operating procedures, surveillance procedures, calibration procedures, current approved calculations, and approved setpoint changes.

- Establishing control and use of setpoint data.

Of the 616 setpoints from the ISD, 70% have been verified with 30% having been dispositioned and entered into the PEDB. Of the 70% that have been verified, no discrepancies were identified establishing a condition of operating the plant beyond its design bases. In addition to the verification, the consistency review of Upper Tier documents is assessing consistency between these documents.

The rate of completion of this action plan continues to be slow and anticipated improvements have lagged due to delays caused by emergent plant issues. This area is identified as still needing considerable improvement to achieve a consistent setpoint control program.

MAINTENANCE PROCEDURE

PROCEDURE REVIEWS

Maintenance procedures are reviewed in accordance with Administrative Procedure (AP-3). This review is to verify that the procedures address and implement the requirements made in the FSAR, Technical Specifications, Quality Assurance and NRC Regulations. New and revised procedures are required to undergo a 10CFR50.59 safety screen in accordance with MCM-4. This review verifies that the design, operation, and function of structures, systems, or components described in the FSAR will not be altered, nor will the changes cause structures, systems, or components to be operated, in a manner that would violate design and license bases requirements.

Following the requirements of MCM-4, approximately 70% of the Maintenance Department's procedures have been completed. The scope of this project includes approximately 318 Preventive/Corrective Maintenance Procedures, 35 Maintenance Directives, and 6 Administrative Procedures for which the Maintenance Department is responsible. The schedule of priority of this project is based on the periodic review dates, Temporary (Term) Procedure expiration dates, and plant scheduling requirements. The remaining 30% are scheduled to be completed by the end of 1997. The Authority is confident that adherence to the processes described in this document provide reasonable assurance that design bases requirements are being properly translated into maintenance procedures and that, if inconsistencies are found, they are evaluated and proper corrective actions are taken.

PREVENTIVE MAINTENANCE PROGRAM

A comprehensive Preventive Maintenance (PM) Program Upgrade was initiated during the RS94 Outage. NRC-Restart Issue NRC- 2 was initiated to review, upgrade and improve the station PM Program. This program included a review of procedure scope/adequacy, scheduling, and confirmation of appropriate vendor technical information for inclusion in the program.

Over six hundred safety related components, from fifteen component groups, which exhibited a higher than industry average failure rate at IP3, were analyzed for PM Program adequacy. A list of 136 components which should have PMs performed was generated as a result of this review. Selection of these components was based on past failure history at IP3. Generic PM recommendations were made for each of these components. Many of these components were already in the IP3 PM Program, but required additional scope. Upon completion of the PM, the work packages were reviewed and evaluated by Maintenance Engineering. Results of these reviews were documented and adjustments were made to the PM Program.

The majority of Vendor recommendations identified during this audit were already incorporated into the PM Program, but at different performance frequencies. Vendors generally accept changes to PM frequencies based on plant experience. A review of plant history indicates that the PM tasks and frequencies in the PM Program are generally adequate.

A database assessment of the PM program was conducted. This involved a review of PM program findings from NRC, INPO and QA. Plant personnel were interviewed and other plants with effective PM Programs were contacted. The goal of this review was to determine whether there were any commitments and/or potential commitments relating to the IP3 component PM Program that were not satisfactorily incorporated.

As a result of this assessment, over 1000 documents identified in numerous databases were reviewed. The IP3 Licensing and ORG databases were searched for issues dealing with PM. The purpose of these reviews was to find commitments that were made which concerned the IP3 PM Program. When a commitment was identified, existing IP3 programs and/or documents were reviewed to ensure that the commitment was implemented.

The IP3 PM Program is controlled by Administrative Procedure AP-55, and includes over 3000 Maintenance Department PM Tasks and over 1500 I&C Department PC/PM tasks. This procedure establishes the elements of an effective integrated PM Program. In order to assess the effectiveness of actions taken to improve the plant PM Program, the following trending is performed:

1. Number of components coming due each month under the present PM Program.
2. Number of PMs actually being performed monthly
3. Number of PMs past due and entering the 125% grace periods
4. Number of PMs overdue by more than 125%

The results of this trending are included in quarterly reports issued to management.

Based on the results of reviews performed, the fact that the IP3 PM program is subject to continuous review, and that frequent changes are made (when appropriate) to improve the quality and efficiency of the PM Program, it is concluded that the IP3 PM Program is adequate in monitoring the performance of, and maintains IP3 Structures, Systems and Components to meet the requirements of the IP3 Design Bases.

SURVEILLANCE TEST PROGRAM

The Surveillance Test Program (STP) controls the scheduling and performance of the Technical Specifications Surveillance Tests.

The scope of the Surveillance Test Program consists of approximately 300 Technical Specification required surveillances, and approximately 300 other requirements from the FSAR, Operational Specifications and ASME Section XI Code. The Surveillance Test Program also encompasses Technical Specification required instrument calibrations.

The program is controlled by Administrative Procedure AP-19, Surveillance Test Program. The program was developed by reviewing Technical Specifications and ensuring that procedural requirements meet Technical Specifications.

NYPA implemented a number of Surveillance Program enhancements that captured the inclusion of additional design feature testing predicated on inputs such as plant modifications, Standard Technical Specification reviews, industry operational experience reviews, and audits / assessments. In the early to mid-1990's NYPA implemented a number of Surveillance Program improvement efforts that were captured by projects such as IP3's Surveillance Test Results Improvement Program (STRIP) (presented to NRC in a January 1993 Management Meeting), Plant Improvement Programs (PIP) items 129.1, 131.1, 151.1, and the Restart & Continuous Improvement Plan (RCIP) Action Plan R-3.1.1.5.

The Authority has conducted two major efforts in the past 5 years that checked that design bases information is reflected in surveillance tests; a 24 month cycle surveillance test extension review, and the NRC Restart Issue #40. The 24 month cycle effort required that IP3 personnel examine existing tests and Technical Specification setpoints to ensure that design bases setpoints would remain valid. It also required a review of the Surveillance Test Program to ensure that Technical Specifications surveillance requirements would be valid during the longer fuel cycle. The 18 Month Technical Specifications surveillance procedures were reviewed in accordance with the guidance in NRC Generic letter 91-04, Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle.

During the plant restart in 1995, a project was performed to ensure Surveillance Program adequacy. This project and results are documented in NRC Restart Issue #40. This project required each plant department to walk-down and validate their test procedures. This process resulted in improvements to the procedures. Also, a feedback process was established which enables test performers to continue validation of the procedures. Specific training was given to test performers on this matter to ensure continuous improvement. This effort also required a validation that the plant's Technical Specifications surveillance procedures Operability Criteria met the intent of Technical Specifications and hence, design bases requirements. A line-by-line comparison of Technical Specifications vs. the appropriate surveillance test was completed and a test vs. Technical Specifications cross matrix was developed.

A review of required calibrations and their associated alarms and trips was performed as a result of a License Event Report in 1993. The review verified that Technical Specification Surveillance Requirements were being performed as per Technical Specifications.

Peer reviews are performed on completed tests by line management personnel in accordance with AP-19. This review is performed to identify discrepancies in the test results and to assess if Technical Specification, Operational Specification or IST operability criteria are satisfied. This review along with the review done by operations is designed to identify any operational issues.

The test program is also used following maintenance of plant equipment to ensure that design bases requirements are still being met. Following maintenance, surveillance test procedures are used as necessary and appropriate to ensure that components are operable. When Technical Specifications surveillance components are maintained, the applicable surveillance test(s) are used or the criteria from the test procedures are referenced in post maintenance work requests, as appropriate. This ensures that design bases information is met following repairs to plant equipment.

The procedures in the Surveillance Test Program are periodically reviewed in accordance with plant requirements. The procedure review process utilizes a review against the FSAR. This ensures that design bases information is reviewed in a periodic fashion. The Surveillance Test Program procedures are presently being reviewed and upgraded in accordance with the Procedure Writing Manual. This entails a process that utilizes an MCM-4 Safety Screen. Since October 1995, approximately 47% of the 606 total tests in the Surveillance Program have been reviewed by Qualified Safety Reviewers utilizing the MCM-4 screening process. Presently, a review of the Reactor Protection System, Engineered Safeguard Systems and Diesel Start System tests as required by Generic Letter 96-01 is underway.

IP3's QA Program periodically Performs assessments to determine that Technical Specification required testing is being performed.

A number of Surveillance program deficiencies which involved calibration deficiencies, proper performance of testing, and technical adequacy of surveillance tests were resolved as part of our RCIP action plan and several NRC Restart Items.

QA AUDITS

In the area of procedures, QA audit activities generally address implementation of Operation, Maintenance, and Testing Procedures. The procedures are reviewed for consistency with NRC Commitments, FSAR, Technical Specifications, Operations Specifications, and Regulatory requirements on a sample basis. The QA organization evaluates the effectiveness of design control activities in meeting 10CFR50, Appendix B (Criterion 3) requirements on a 24 month basis for Design Control as required by Technical Specifications.

The following are examples of recent inspection summaries describing the results of evaluations regarding IP3's effectiveness in translating design bases requirements into operating, maintenance, and testing procedures:

First Quarter 1995 Performance Assessment and Trend Report

A program deficiency was identified and documented by Site Engineering. The finding identified that modification installation tests were improperly performed due to tests being incomplete, improperly written or improperly executed. As a result of this finding a task force was established and reviewed the post modification testing process. Approximately 500 modifications, minor modifications, and design changes were reviewed. As a result of this review, the following actions were taken or initiated:

- Testing standards utilized for post maintenance testing were provided to the post modification test group for use when appropriate.
- Training, identifying the types of deficiencies found, was provided to engineers responsible for preparation of test procedures.
- Engineering assurance personnel review the quality of a sample of new test procedures and provided feedback to the Engineering Department.
- Enhanced procedural guidance on post modification testing requirements was developed and implemented.

Third Quarter 1995 Performance Assessment and Trend Report

The reactor was operated at a pressure lower than analyzed in the FSAR. The following two reviews were part of the required corrective actions associated with this event:

- Headquarters Reactor Engineering performed a review which compared a selected scope of operating procedures against assumptions made in the Accident Analysis Basis Document (AABD). Training on the Core Operating Limits and assumptions used in the accident analysis of IP3 for Design Bases Events was given to Plant Operations Review Committee members, plant operators, and managers and supervisors of technical groups. Similar training is being given to Engineers as part of the engineering Support Personnel (ESP) program. This issue is discussed in more detail in the Operating Procedures section.

- The Independent Safety Engineering Group (ISEG) at NYPA conducted an extent of condition review of Operating Procedure revisions. Additionally, training was conducted for plant staff to provide a better understanding of the plant's design bases. This issue is discussed in more detail in the Operating Procedures section.

Fourth Quarter 1996 Integrated Self Assessment / Trend Report

In the area of Engineering and Technical Support the following issues were identified;

- The Document Change Resolution backlog has decreased significantly during the quarter and is below the station goal.

- The Setpoint Control Program has continued to experience schedule slippages as resources are redirected. New personnel have been assigned to this effort to assist in the processing of setpoint open items.

- The annual Technical Specification audit of the Surveillance Test Program was conducted by Quality Assurance in December, 1996. The audit verified that the implementation and control of the program is satisfactory and continues to improve. The performance of test personnel observed during test activities was considered excellent. However, several procedural inadequacies were identified, as well as difficulties in translating technical requirments into programs and processes. The need for continued improvement in surveillance test procedures remains. One problem area related to inadequate reviews of existing procedures when new Operational Specifications are issued.

CONCLUSION

Indian point 3 has procedurally controlled programs in place which define the processes for translation of the existing plant design bases, and changes to the design bases, into the plant operating, maintenance, and testing procedures. The strength of these processes is based upon the multiple level of qualified reviews of new and revised procedures, and the tracking mechanisms for followup and incorporation of design changes into plant documents.

The existence of the programs described, the specific efforts focused on upgrades and validations, and the actions for continued improvement identified both internally and externally, provide the rationale for NYPA to conclude with adequate confidence that design bases requirements are translated into plant operating, maintenance, and testing procedures.

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

During the life of IP3, the Authority has established processes and undertaken many initiatives to provide reasonable assurance that SSC configuration and performance are consistent with the plant's design bases. Confidence in the adequacy of these processes is based on a multi-level approach which includes management, control, and verification of the configuration and performance of SSC. The processes and initiatives that provide this confidence include:

- Configuration, Design, and Document Control processes
- Design Bases Document Program
- Programs, Walkdowns, and Reconstitution Activities
- Quality Assurance Audits, Management Oversight, and Self Assessments
- Testing (Post Modification, Post Maintenance, and Surveillance)
- INPO Assessments and NRC Inspections

Confidence is also based on the quality of work, the training and performance of its personnel, management expectations on procedure use and adherence, and the generally positive results of assessment activities.

The Authority uses programs and processes, such as those listed above, to continuously evaluate system, structure and component configuration and performance. If deficiencies are identified they are evaluated and corrective action taken as described in section (d) of this attachment.

The configuration, design, and document control processes are discussed in section (a) of this report. Design Bases Documents are discussed in section (f) of this report. Discussion of other areas is provided below.

PROGRAMS, WALKDOWNS AND RECONSTITUTION ACTIVITIES

In establishing and maintaining consistency of SSC configuration and conformance to design bases, the Authority has undertaken several initiatives. These initiatives include design document reconstitution, design reviews, field walkdowns, verifications and programmatic activities.

The following describes significant initiatives conducted to validate design information against design bases requirements.

ELECTRICAL AREA

Electrical Distribution Systems (EDS) programs and processes have been completed or are currently in use to maintain consistency of electrical system and component configuration and performance with the design information.

To facilitate control of EDS changes, a verified and validated Electrical Distribution System Model (EDSM) was developed in 1992. This model consists of the EDS data such as loads, transformer, motor and cable impedances, cable sizes and lengths, fuse and breaker characteristics, as well as calculational modules to calculate short circuits, voltage drops, bus loading and fuse and breaker coordination. This model facilitates update of affected EDS calculations, studies and analyses.

The Electrical Distribution System Model includes a series of calculations covering various plant normal and accident operating modes specifically for the 480VAC Safeguards Buses and safety related Motor Control Centers. The calculations analyze load flow, voltage drop, and short circuit conditions and breaker/fuse coordination. This evaluates the ability of the 480V safeguards system to successfully perform its safety related functions during design basis accidents, and during normal plant operation.

Calculations for the safety related 125 volt D.C. busses and the 120 volt A.C. instrument busses are not included in the computer model. These calculations were reconstituted and exist as the manual calculations.

Loading calculations were developed with input from the Operations Department using operating procedures such as Emergency Operating Procedures (EOPs). Operating procedures were revised to reflect calculational results.

Electrical Distribution System (EDS) changes are required to be submitted to the Electrical Design organization for review of proposed changes to determine their effect on the EDS configuration, and to evaluate potential effects on the design bases. In addition to this review, the design process requires development of any required calculations and analyses and/or revisions of the affected design calculations including, but not limited to loading studies, electrical short circuit studies, coordination studies and voltage studies.

The Electrical Distribution System Model (EDSM) described above forms a key part of the overall IP3 Design Bases Reconstitution effort. The calculations in the EDSM include the design bases of the postulated worst case plant conditions thereby establishing a limiting design baseline. A limiting baseline is an operating limit that the plant is not permitted to exceed and is dependent upon the plant configuration. For example, the 480VAC Safety Buses are not designed to operate at voltages higher than 500 volts. The calculations have shown that operation at voltages higher than 500 volts could potentially occur during the Cold Shutdown mode of plant operation if voltage and loading are not controlled. Consequently, procedural limits, which include specific operator actions, have been established to preclude this limit from being exceeded.

Calculations for the acceptability of the proposed change are finalized prior to the completion of final design. Following completion of the proposed EDS change, the calculations are to be entered into the Electrical Distribution System Model and 125 volt D.C. and 120 volt A.C. calculations are updated to reflect the plant change.

As a result of a QA audit in this area, weaknesses were identified on the timeliness of the model update. These weaknesses are presently being addressed through the corrective action process.

Detailed assessments and walkdowns were performed in support of the: EDS reconstitution process; updating selected EDS data such as the fuse, cable, motors, transformers; and to re-verify compliance with design bases. Other walkdowns were performed as a result of lessons learned or as part of verifying the extent of condition for identified deficiencies. Following are examples where field conditions were verified against design documents:

Cable Separation

Field walkdowns were conducted to assess that the separation and channelization of cables and trays for compliance with the design bases (FSAR Sections 7.2, 8.2 and 8.4). These walkdowns and associated assessments consisted of a cable tray separation phase and a cable channelization phase.

The cable separation issue was identified in 1991. Engineering performed cable tray and fire barrier walkdowns in 1991 and 1992. The walkdown reviewed the conformance of cable tray and exposed cable path separation to the plant's design basis criteria contained in FSAR Section 8.4, and consisted of measuring distance between trays, exposed cable paths and penetrations of different channels. Issues identified were documented in Licensee Event Reports (LERs) 91-08, 91-08-01, 92-18 and NRC Inspection Reports. These issues and discrepancies were evaluated and corrected as required.

The channelization phase walkdown was an extension (extent of condition) of the cable separation phase which looked for cross channelization to verify compliance with the single failure criteria contained in the FSAR. This phase commenced in 1993 and consisted of walkdowns of selected Category I cable routes in the Control Building, Auxiliary Boiler Feedpump Building, Electrical Tunnels, Turbine Building, Primary Auxiliary Building and the Containment Building. LER 93-025 documents the only reportable channelization deviation which resulted from this walkdown.

Initiatives undertaken to prevent future cable/tray separation deficiencies include training of personnel performing installation, review of design changes for cable separation, and surveillance of tray fire barriers. The Authority currently uses an improved computerized cable and raceway data management system which performs cable channelization/separation automatically in addition to performing calculations on raceway fill, weight monitoring, voltage class and safety class separation.

Fuse Control

Field walkdowns of safety-related 480 volt A.C, 125 volt D.C. and 120 volt A.C. fuses to record their type, size and rating were performed in 1992. The field data was compared with the existing design documents and discrepancies were resolved.

Loading and coordination calculations were re-done to determine that the installed fuses were applied correctly and, if not, replacement fuses were installed.

A controlled master fuse list was established as well as procedures to control the selection and replacement of fuses. The procedures require fuse replacements to be authorized by the Electrical Design organization.

CIVIL / STRUCTURAL AREA

Seismic Analysis

In response to I.E. Bulletin 79-14, a field verification program was implemented in accordance with the requirements of this Bulletin, to determine that the drawings and specifications utilized in the seismic re-analysis represented the as-built configuration. Line walks of 194 static "span table analyzed" lines and 117 "dynamically analyzed" lines were performed. Quality Control documentation verifying compliance with bolt design requirements was identified. Verification surveys involving hundreds of base plates and anchors were conducted to verify that the design and installation met design requirements and bases. This effort also involved ultrasonic testing to determine anchor embedment measurements and torque testing for anchor bolts.

Systems Interaction Study

In 1982 the Authority performed a Systems Interaction Study at IP3 in response to an Advisory Committee on Reactor Safeguards (ACRS) recommendation. The objective of this study was to establish confidence that the plant structures, systems and components would not be prevented from carrying out their safety functions by interactions with non-safety related structures systems and components when subjected to any design basis initiating event. An interdisciplinary team of experienced engineers performed structures and system walkdowns to identify potential interactions. Approximately 6000 potential interactions were identified by the team. On further review 90% were found acceptable leaving 643 open potential interactions to be resolved. These open interactions were closed out by further walkdowns, design reviews, analyses, tests and modifications.

SQUG Program

In response to NRC Generic Letter 87-02 / USI A-46 (SQUG Program), the Authority performed reviews, evaluations and walkdowns to verify compliance with the applicable criteria and guidance. About six hundred components were identified and walked down for seismic verification by a team of qualified individuals. This program identified sixty seven (67) mechanical and electrical equipment outliers requiring resolution. All outliers were reviewed to determine compliance with design documentation, and when deviations were found the plant procedures were followed to identify and resolve the noted conditions. None of the outliers were found to violate the design basis. To date, 27 of the 67 outliers have been resolved. The remaining are scheduled to be resolved no later than startup from refueling outage reload 10/cycle 11 (1999).

Bolt Thread Engagement Walkdowns

As part of an extent of condition, walkdowns were performed in 1993 on portions of Service Water, Component Cooling Water and Residual Heat Removal piping systems to identify potential bolting thread engagement non-conformance issues. A total of 290 bolted piping connections were inspected. Twenty-two deficiencies were identified and subsequent analysis established that connections were consistent with the design information.

Branch Life Connection Investigation

In 1993 the Authority performed an investigation of the IP3 branch pipe connections for erosion/corrosion. This included design document reviews, evaluations, analyses, visual inspections and non-destructive examinations to measure pipe wall thickness. This investigation included Non-Category I Systems with design pressure greater than 100 psig and design temperature greater than 162 ° F. A total of 305 branch connections were evaluated by this program. Fifteen connections involved non-conformance with ANSI B31.1 and corrective actions were completed.

Inservice Inspection Program

The integrity and functionality of welds and supports in ASME Code Class 1, 2, and 3 systems are checked under the Inservice Inspection Program in accordance with ASME Section XI Code. A set of ISI flow diagrams showing Class 1, 2 and 3 piping and components is maintained and updated regularly. Systems and components which are not code classified, but are considered important to safety, may come under the heading, "Augmented Inservice Inspections," and included in the ISI program. Augmented inspections may also include commitments made to the Nuclear Regulatory Commission, or the examination of code classified systems more frequently than required by the code.

MECHANICAL AREA

HVAC

In 1992 the Authority initiated an effort to consolidate the Heating, Ventilation and Air Conditioning (HVAC) design information. This effort included retrieval and organization of design records, reconstitution of missing design documents and confirmation that plant testing, maintenance and operations are within the design and licensing bases.

The heating and ventilation of some areas of the plant containing safety-related systems and components, i.e., the Control Building, the Primary Auxiliary Building, the Fuel Storage Building and the Emergency Diesel Generation Building were re-evaluated. HVAC calculations for the above building areas were updated or reconstituted. Original plant design deficiencies as well as the effects of new heat loads, were identified as a result of this effort and corrective actions were developed and implemented. These corrective actions included the upgrading of the QA classification of certain HVAC equipment, re-powering of ventilation fans from alternate power sources, and addition of temperature monitoring instrumentation.

The addition of new heat loads as well as the increase of the river water temperature limit to 95°F were also incorporated in this re-assessment.

Modifications were implemented in several plant areas to increase the capacity of HVAC systems and to improve their flexibility.

Inservice Testing Program

The Inservice Testing (IST) Program at IP3 includes specific testing requirements for pumps and valves performing specific safety-related functions. The results of such tests are used in assessing operational readiness of certain pumps and valves.

The Inservice Testing Program tracks and trends the values obtained for valve and pump performance parameters. These valve and pump performance parameters are used to determine if the subject valves and pumps are operable. The operability criteria used in the IST program are developed from the FSAR, TS and Section XI of the ASME code.

The IST program verifies that the pump and valve performance criteria as specified in the FSAR are maintained. Deficiencies are addressed by the corrective action program discussed in section (d).

Erosion / Corrosion Program

The Erosion/Corrosion Program is a monitoring and inspection program that predicts and verifies pipewall thinning at IP3 as required by I&E Bulletin 87-01.

The erosion/corrosion program is used to check that the piping systems remain operable and do not pose a personnel safety concern or require an unscheduled plant shutdown as the plant ages.

This program at IP3 consists of two parts, the Large Bore and the Small Bore program.

The Large Bore Program uses a computer program which predicts wear rates in modelled piping systems based upon flow rate, pressure and temperature of the media, plant chemistry, plan run time, and piping geometry. The components that show the most severe wear are then inspected ultrasonically. Results of the ultrasonic inspection are inputed into the computer program to refine the model and ensure timely pipe replacement and/or greater accuracy in the next predicted wear rates.

The Small Bore Program predicts the susceptibility of small lines (2" and smaller) using the methodology of the large bore program, but without a computer model. Input from Operations personnel, review of maintenance records, and application of engineering judgement are also used to predict the susceptibility of each line. The consequences of failure of each susceptible line are also considered. Results of the small bore inspections are evaluated and used in determining the priority of the next outages inspection points and priorities.

Piping components are replaced as necessary when actual wear approaches code calculated limits.

Motor Operated Valve (MOV) Program

In response to Generic Letter 89-10 a program was developed to: establish and document the design basis for each safety related motor operated valve (MOV); develop calculations to confirm that each such MOV is capable of meeting its design bases requirements; and test each MOV in a manner that confirmed the calculational methodology and MOV operability.

The program looked at such factors as design requirements (delta-p flow, process temperature, ambient factors, etc.) to establish the design conditions that each safety related MOV must operate under. Applicable information contained in the UFSAR, Technical Specifications, DBDs, System Descriptions, Operating and Test Procedures were reviewed for the purpose of fully establishing the hydraulic and environmental conditions under which a given MOV would be required to perform its safety function. This information was documented in a specific calculation (DP Calc) for each valve or valve group.

Work on the IP3 GL 89-10 program was initiated in 1990 and is continuing through the present under a commitment to the NRC for program completion within 60 days following startup after refueling outage 09 which is scheduled for the Spring of 1997.

The following design basis reviews were conducted for each IP3 GL 89-10 MOV:

Design basis reviews documenting the functional requirements for all eighty-nine (89) MOVs included in this program have been completed. This includes the maximum expected differential and line pressures, flow rates and fluid and ambient temperatures, degraded voltage conditions and maximum allowable open and close stroke times.

The evaluation was based on review of the pertinent design documents, FSAR, Technical Specifications, normal and emergency operating procedures, associated determination of worst-case cycling scenarios within the design basis, system flow diagrams and electrical wiring and logic diagrams.

Assessment of the MOV design characteristics and bases (i.e., valve, operator, motor) have been completed for all program MOVs. Verification included:

- Review of design specification/installation requirements
- Cable sizing, routing, separation including assessment of control circuit function/design.
- Seismic/mounting requirements (operator orientation)
- Licensing issues affecting installation
- Electrical/mechanical interlocks (indication and control)
- Field walkdowns/inspections to aid in verification of MOV assembly (e.g, nameplate data, wiring configuration, valve/operator/motor orientation, etc.)
- Documentation of discrepancies identified as part of MOV assembly design verification were initiated for resolution under the DDOI or DER system.

Maintenance Rule Implementation

Continued assurance that Structures, Systems and Components (SSCs) will function consistent with Design Bases requirements is provided in part by the Maintenance Rule Program implemented pursuant to 10CFR50.65. The Maintenance Rule Program implementation consists of two major activities:

1) Establishment of baseline program
2) Continuing monitoring program

The baseline program determined which SSCs should be included within the scope of the maintenance rule. The functions of plant systems were identified by Systems Engineers from various sources. These functions were presented to a multi disciplinary group (Maintenance Rule Expert Panel) for review. Scoping criteria and guidance provided in NUMARC 93-01 guideline was used to determine the systems and functions included in the Maintenance Rule Program. The criteria are as follows:

1) Safety related SSCs
2) Non-safety related SSCs that mitigate accidents or transients
3) Non-safety related SSCs that are used for Emergency Operating Procedures (EOPs)
4) Non-safety related SSCs whose failure prevents safety related SSCs from fulfilling their safety-related function
5) Non-safety related SSCs whose failure causes scrams or actuates safety systems

Technical Specifications, FSAR, DBDs, Normal and Emergency Operating Procedures, Probabilistic Risk Assessment/Individual Plant Examination (PRA/IPE) reports, Industry Operating Experience information including LERs and System Engineer reviews were all used as sources of information in the scoping effort.

When a function met any one of the scoping criteria listed above, then its associated system, subsystem, trains, subtrains, groups of components that support in scope functions were included to be in the scope of the Maintenance Rule.

Performance criteria commensurate with risk (as determined by IPE/PRA and the Maintenance Rule expert panel) were established to effectively monitor performance of system functions within the scope of the Maintenance Rule. Plant level performance criteria are used to monitor selected functions in the scope of the Maintenance Rule. Where this was not sufficient, system and train level performance criteria were developed. Performance criteria are continually monitored to increase SSCs reliability and availability.

FIRE PROTECTION / APPENDIX R

Re-evaluation of the Appendix R Safe Shutdown Analysis

In 1994 the Authority re-visited the IP3 Appendix R Safe Shutdown Analysis to review continued compliance in light of current industry issues and practices and to resolve identified weaknesses. Modifications installed since the 1984 analysis were reviewed to determine their impact on Appendix R.

Fire Barrier Inspection

A walkdown and baseline inspection of IP3 fire barrier penetration seals was performed in 1993 via ENG-527 Fire Barrier Inspection. This effort included data acquisition phase and an evaluation and review phase.

Each barrier was inspected in a systematic manner to ensure that each penetration was identified. Each penetration and its associated fire seal was then inspected against specific inspection criteria based on the type of seal material installed. In addition to the inspection of each penetration, conduits were inspected to determine the need for their sealing.

The inspection criteria were intended to identify deficiencies such as unacceptable holes, separation or shrinkage in the surface of each seal. The results of the inspection of each seal surface were combined and the overall fire seal configuration was considered. The review included specific information such as condition of each side (integrity), the overall depth of the seal, the assigned typical seal detail, the referenced fire test and tested configuration, additional testing as necessary, etc. As a result of this review, each fire seal was judged to be functional or non-functional.

Of the total number 1200 fire seals inspected, 8% of the seals were judged to be non-functional with the remaining 92% judged to be functional. All non-functional fire seals were repaired.

Review of the plant fire protection features for compliance with the appropriate NFPA Codes

In 1993 the Authority completed a review and walkdown of IP3 fire protection system components and features for compliance with the National Fire Protection Association (NFPA) code of record. The FSAR was initially reviewed for specific commitment to codes and code editions. The objective of the NFPA Code Walkdown effort was to: (a) verify that the Authority is in compliance with the NFPA Code of record, (b) identify any non-conformances, determine their effect on operability, and (c) provide engineering evaluations and recommend any corrective actions regarding non-conformances.

The NFPA Code Walkdown effort, which is a line by line compliance review, included: (a) preparation of a code shell for each code section, (b) review of all applicable drawings and documentation, (c) walkdown of field conditions, (d) identifying any non-conformances, (e) identifying impact on operability of any non-conformances, (f) review of surveillance testing, operation and maintenance procedures for code compliance (g) preparation of justification for deviations, if applicable, and (h) tracking to closure items that require procedure or plant changes.

Identified NFPA code non-conformances were evaluated by the system engineer to determine the need for a DER and subsequent operability review. If a DER was not warranted, the non-conformance was evaluated by a fire protection engineer either as "acceptable as is" with technical justification provided, or included in a non-conformance matrix requiring additional work and tracking to closure.

The impact of these non-conformances was evaluated immediately upon identification. Final closure of identified non-conformances is in progress.

Development of Hydraulic Calculations

Since hydraulic calculations for several water-based suppression systems could not be located, the Authority is in the process of developing a hydraulic model. This hydraulic model will serve as a tool to determine system wide effects of additions or modifications to the fire protection systems. Guidance and requirements for performing and controlling hydraulic calculations are provided in FPES-06, Hydraulic Calculations for Fire Protection Systems. Currently hydraulic calculations for the Fire Protection System have been performed and are in the review and approval process. This effort is expected to be completed by April 1997.

Development of the Fire Protection Design Basis Document

The latest design information is contained in the IP3 Fire Protection Design Basis Document that was completed in October of 1996. The preparation of the Fire Protection Design Basis Document included verifying that the applicable portions of the IP3 FSAR are correct as well as correctly implemented in the field. Proposed changes to the FSAR resulting from the latest Appendix R Safe Shutdown Analysis Report are being evaluated and resolved.

SYSTEM ENGINEERING PROCESS

The primary objective of the System Engineering Process is to improve plant safety, efficiency and availability by improving and monitoring system health through timely involvement in the identification and resolution of system problems and by ensuring that systems are operated and maintained in accordance with design basis requirements.

Each System Engineer has the responsibility for their assigned risk significant and selected systems to monitor system health and performance using: System walkdowns, trend analysis, Problem Identification Descriptions (PIDs) and Work Request (WRs), Action Commitment Tracking System (ACTs), Deviation Event Report (DERs), and industry operating experience. System Engineers assist the Operations Department by taking a lead role on system-related emergent issues and remain cognizant of component-related emergent issues.

System walkdowns allow the system engineer to assess system status, material condition, monitor plant cleanliness and radiological conditions. This is accomplished by identifying safety hazards, equipment problems or discrepancies. These walkdowns are intended to supplement, not replace, operability walkdowns performed by other organizations, such by the Operations Department. During system walkdowns, specific consideration is given to identifying non-standard or undocumented modification installations, whether temporary or permanent in nature. The results of the system walkdown are documented in the form of a System Engineering Report. These reports are copied to the Operations Manager and results are summarized in the quarterly System Engineer Status Report (per TSP-53-"System Performance Monitoring and Trending"). Deficiencies noted during a walkdown are documented and resolved.

OTHER INITIATIVES

System Certification Walkdowns

System Certification Walkdowns were performed as part of the restart effort in 1995 in accordance with the System Certification Program Plan. Seventy-four systems were included in this process. The System Certification Program served the role of assessing the material condition and functionality of selected plant systems and their ability to support the safe restart of the unit and sustain unit reliability throughout the operating cycle. The program was centered upon a joint System Engineering and Operations assessment of the system's material condition, configuration and readiness. The intent of the process was to confirm that (1) system hardware was in place and consistent with the physical configuration described in the system's design documentation (2) system hardware was in acceptable physical condition and that discrepancies were appropriately evaluated and (3) that the system reliability demonstrated the functional capability described in the FSAR and IP3 Technical Specifications. This effort also included review of open Work Requests, Problem Identification Descriptions (PIDs) and Deviation Event Reports (DERs) to determine the readiness of systems for startup. The system certification process focused on physical conditions, system functionality, verification of system configuration with the design bases through walkdowns utilizing detailed plant drawings, and system reliability as determined by surveillance testing and historical performance. Walkdowns included comparison of the physical plant to control room drawings.

Surveillance Testing Program (STP)

Surveillance testing to verifies the operability of systems and components and ensure variables are within specified design and licensing bases limits. Surveillance requirements are obtained from Technical Specifications, ASME Boiler and Pressure Vessel Code, Section XI Pump and Valve program, ISI/IST requirements, and other criteria as necessary. Programs are in place to assure failed or missed surveillance are appropriately evaluated for reportability and corrective actions are implemented as necessary.

Post Modification and Post Maintenance Testing

The Authority maintains a post modification testing program to verify that plant modifications meet design and operational requirements and criteria. The Modification Procedures require identification and performance of required testing to demonstrate functionality and operability of installed modifications.

Testing is also used following maintenance of plant equipment to ensure that design basis requirements are still being met. Following maintenance, surveillance test procedures are used as necessary and appropriate to ensure that components are operable. If surveillance procedures cannot be used, then separate test procedures are developed. Basically, whenever TS surveillance components are maintained, plant procedures require that the applicable surveillance tests are used or the criteria from the test procedures is referenced in post maintenance work requests. This ensures that design basis information is met following repairs to plant equipment.

Drawing Update

This is an ongoing program and it involves:

- Update of drawings affected by design changes, and
- Update of drawings or other design documents following resolution of configuration / design discrepancies. Resolution of such discrepancies is based on FSAR and plant design bases reviews.

In the early 1980's the Authority initiated a drawing update program. This program involved plant walkdowns to determine that field conditions were correctly reflected in plant drawings, identification of discrepancies and corrective actions.

Presently the Drawing Update Program involves updates of plant drawings to incorporate changes made due to plant modifications or document discrepancies as identified via a Request for Document Change (RDC).

Control room drawings identified by the Operations Department that are vital for the safe operation and shutdown of the plant are annotated to reflect current plant configuration following a modification and are required to be formally updated within 30 days, and within 2 full working days for a temporary modification.

QA and NRC Inspections have in the past identified weaknesses in this areas relative to large drawing update backlog and in the timeliness of control room drawing update. As of January 1996 the backlog of control room drawings requiring update has been reduced and remain at zero. Also, the backlog of drawings requiring update as a result of modifications has been reduced from 15,000 in 1995 to 2000 today.

Commitment Review

To determine if NRC commitments as summarized in the Licensing Commitment Database are satisfactorily addressed, resolved and properly incorporated into plant design, the following effort was undertaken in 1993:

- Review of compliance, completeness and resolution of a sample licensing commitments.
- Performed QA Audit 93-07 (3/93) - Licensing Commitment Review, R.G. 1.97 and QSPDS.
- Performed QA Audit 93-15R (4/93) - Statistical Sample of all Commitments in IP3 Licensing Database.
- Performed QA Audit 93-21R (6/93) - Low Temperature Overpressure Protection System.
- Performed QA Audit 93-22R (10/93) - NUREG-0737, Clarifications of TMI Action Plan Requirements.
- Performed QA Audit 93-29R (9/93) - 10CFR50 Appendix R Requirements (Sections III.3,J,L and 0).

Typical findings identified during the above audits involved: a case of not performing a committed revision of a surveillance procedure on schedule, not completing a modification during the committed refueling outage, not locating documentation to verify committed PORV and Block Valve stroke testing as well as not resolving strip chart recorder problems in the control room.

All identified finding have been documented and resolved.

Ultimate Heat Sink Analysis

An analysis was performed by Westinghouse in July 1989 for a maximum Ultimate Heat Sink temperature of 95°F. The analysis addressed the response of all affected components during normal and post-accident operation for Service Water System (SWS), Component Cooling Water System (CCWS), Residual Heat Removal System (RHRS) and the Spent Fuel Pool Cooling System (SFPCS). Plant procedures were reviewed by Westinghouse and recommended changes were implemented. The changes made in these revisions included setpoints and valve positions and affected procedures. In July 1995 Reactor Engineering conducted an independent audit of the procedural revisions at Indian Point 3 to support the Ultimate Heat Sink Analysis and concluded that design bases as well as recommended requirements had been adequately translated and implemented.

Steam Generator Replacement

In 1989 the Authority replaced the IP3 Steam Generators. In implementing this significant project the Authority investigated and reviewed the design bases of the areas of the plant that were affected by this project. Such areas included the following:

- The design bases and performance requirement of the "old" steam generators were reviewed to determine the design fabrication and installation of the "replacement" steam generators.

- Development of a Nuclear Safety Evaluation and update of the FSAR.

<u>Operator Work Arounds</u>

As described in part (d) of this submittal, the Problem Identification Description (PID) process is used to identify known or suspected problems or deficiencies with plant structures systems and components.

PIDs affecting normal plant configuration that require compensating manual operator action or augmented surveillance or altered operator response are coded as Operator Work Arounds (OWAs).The Operations staff reviews OWAs and prepares a monthly report to plant management. A periodic review is also performed by Operations to assess the aggregate effect of all OWAs on the ability to safely operate the plant.

System Engineers are also responsible for regular assessment of the health of their systems. Procedure TSP-53, "System Performance Monitoring and Trending," provides the guidance needed to perform system monitoring and trending. Open PIDs, including OWAs, are reviewed and assessed during this process.

<u>Labeling Program</u>

This program was developed based on NUREG-0700, EPRI NP-6209 and INPO 88-009. The process requires reviews of design drawings, FSAR drawings and design information. Any deviations are then translated to any associated procedure and Check Off Lists.

<u>Environmental Qualification (EQ) Program</u>

This program addresses the requirements of 10CFR50.49 "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." The requirements for the program are addressed in FSAR Appendix 6F which invokes the requirements stipulated in 10CFR50.49 for compliance and provides the specifics for design bases for environmental qualification. The program incorporates NRC Regulations, Regulatory Guides and other positions and guidelines, as well as IEEE Standards and sound conservative engineering practices to form a detailed and working program to achieve and maintain electrical and post accident monitoring equipment environmental qualifications for Indian Point 3 in accordance with 10CFR50.49 required, licensing bases, and all formal commitments. The EQ Program had developed and maintains auditable qualification files for 10CFR50.49 and DOR Guideline equipment. These files provide support and demonstrate the qualification of EQ equipment to plant specific design requirements.

As a result of the EQ Maintenance Program, the Preventative Maintenance Programs are organized to repair or replace items at regular intervals per applicable procedures and schedules that are necessary to maintain environmental qualifications of equipment.

The current IP3 EQ Program is in the process of being enhanced and to also include procedural improvements. The overall compliance with the NRC requirements as per 10 CFR 50.49 are being achieved, however there are areas of documentation weakness that are being addressed so that compliance can be readily demonstrated.

A self initiated review has resulted in the development and implementation of an action plan for the EQ program enhancement. As part of the enhancement effort, IP3 has become the pilot plant for implementation of the EPRI developed, state of the art electronic environmental qualification program.

Design Document Collection

In 1990 the Authority initiated a document collection effort to ensure the design basis documents were under NYPA's control. The program included an activity to systematically assure that NYPA has design documents that are controlled. This involved collecting documentation from the Authority's A/E's and prime / pub-vendors. This effort included: contact and interface with the A/E's or Vendors to physically view archives and obtaining and indexing design basis information. Documents collected must be reviewed in accordance with Design Control Procedure (DCM-11), for applicability and status, and maintained and updated by NYPA. These documents are indexed and cross referenced to their plant components, systems and structures. The effort began with a document turnover from the major A/E's. NYPA retrieved post-operation and pre-operation documents. Currently, the IP3 database (excluding drawings) has over 17,500 documents.

A request was made to three of 30 vendors to submit a listing of all calculations submitted to the Authority from January 1, 1992 to June 1995. The list of calculations submitted by the vendors was compared to the controlled document database to determine if any were missing. It was determined that 41 calculations were missing from the 284 calculations identified by the vendors. All 41 calculations were obtained from the vendors and have been subsequently reviewed for acceptance. It was determined that the sample population was too limited and Revision 5 of the Action Plan increased the scope of the sample to all the vendors who submitted calculations to NYPA for the same time period. This task is currently in progress and is scheduled to be completed in April 1997.

A review of 1,863 references listed in the Design Basis Documents (DBDs) against the controlled document database resulted in 21 calculations that were determined to be missing. These calculations were obtained and reviews for acceptance completed. An effort to review the reference sections of DBD revisions is ongoing to ensure that NYPA has the documents referenced in the DBD's.

A review of the calculations completed by vendors through the modification process was checked by performing a sample of completed and in progress modifications. Thirty three (33) modifications were sampled (10% of the modifications from January 1992 to July 1995). Fifty one (51) calculations were found in various modification packages, all were indexed in the controlled document database. This effort is complete.

Setpoint Control Program

In response to a need for improvements in controlling setpoint values, the Setpoint Change Request (SCR) process, as governed by Modification Control Manual procedure MCM-8, was established in 1993. This process is used to provide configuration control of setpoint values in accordance with Appendix B of 10CFR50. Further details of this program are provided in sections (a) and (b) of this report.

QA AUDITS, MANAGEMENT OVERSIGHT AND SELF ASSESSMENTS

Quality Assurance Audits

Design Control audits, performed bi-annually by the NYPA Quality Assurance Department, reinforce configuration and design bases programs and activities by providing feedback on their effectiveness by identifying weakness and overseeing corrective actions.

In addition, other audit activities review plant configuration, for major programs such as fire protection/Appendix R, security, etc., are conducted by the QA organization, with technical support, as required.

Such audits verify if:

- Applicable operating, maintenance and test procedures covered by audits are reviewed.
- Applicable FSAR sections, Technical Specifications, Operational Specifications and other regulatory requirements are addressed.
- Commitments made to the NRC are implemented.
- Deviations between FSAR and configuration deviations are noted and corrective actions are pursued.
- SSC configuration and performance are consistent with the design bases.

Examples of specific design control audits and associated results are provided below.

<u>Audit 94-11 (6/94)</u>

This audit reviewed the design activities performed at IP3, JAF and in the WPO to support each of the plants. In addition to the aspects of the audits described above, this audit also addressed previous audit findings and weaknesses related to overall design and the specific modification process. Specific programs such as Setpoint Control, Fuse Control, Electrical System Distribution Baseline, DBD, etc. were not covered in this audit since they were scheduled to be individually evaluated in 1995 under specific audits.

The audit concluded that design activities comply with 10CFR50, Appendix B, Criterion III (Design Control) requirements and applicable procedures.

This audit also identified weaknesses in the areas of corrective action, change control and in the timeliness of review of potential plant deviation.

Issues identified during this audit have been addressed by plant management and audit findings have been closed.

<u>Audit A96-04W (6/96)</u>

The purpose of this audit was to assess the effectiveness of Design Control activities at IP3 and compliance with 10CFR50, Appendix B for the following areas:

- Electrical Load Distribution related calculations

- Fuse Control
- Seismic qualification
- Control of calculations
- Post-modification testing
- Engineering Change Notices (ECN)
- System Engineering responsibilities, including Maintenance Rule, preventive/predictive maintenance, tracking system health
- Management assessment of work planning and scheduling, backlog reduction

The audit concluded that the design control process is being adequately implemented. Design documents were being complied with and the overall design bases were being maintained. Areas which were identified as problem areas in the past such as control of calculations, post-modification testing and modification closeouts have shown improvements. Deficiencies were identified in the areas of training and qualification (use of personnel who were not qualified, and a knowledge deficiency in pressure testing requirements); control of modifications (declaring mods operable with unapproved calculations, no formal design verification for temporary modifications and procedure calculation methodology); and attention to detail issues (administrative deficiencies). Two specific areas were identified to require continued management focus, engineering work planning and scheduling and System Engineering.

During this audit a review of DERs affecting design control activities over the past three years revealed that numerous DERs were issued for specifics of the design control aspect of the audited activity. None of these DERs which were issued were classified as significant or programmatic in nature.

Previous audits and Deviation and Event Reports had identified a significant number of fuse related problems. Most of these problems involved Master Fuse List errors and installed fuse discrepancies. The fuse control issues and the current fuse control program were reviewed during this audit. The audit team noted that strict implementation of this program will prevent recurrence of problems.

Management Oversight and Self Assessments

In addition to Quality Assurance audit and surveillance, management oversight as well as self assessment activities and inspections provide reasonable assurance that activities at IP3 are in conformance with the licensing and design bases requirements and commitments.

Following is a description of such programs and activities:

<u>ISEG</u>

Although IP3 is not licensed as a facility which requires an Independent Safety Engineering Group (ISEG), implementation of a non-licensed based organization by the same name was initiated in September of 1994. The IP3 group has experience in engineering and operations and consists of an ISEG Director based in the corporate office and a Senior Assessment Engineer at the site.

ISEG performs independent reviews of selected nuclear safety evaluations and plant activities to identify problems that may have an adverse impact on plant safety and to assess if the plant is being operated in accordance with the design basis. These reviews include periodic walkdowns of control room panels and other areas to observe indication of plant parameters and operating conditions.

Engineering Assurance

An Engineering Assurance and Self Assessment Program was established in early 1995 within the Design Engineering Division to strengthen the oversight and control of the design engineering process.

An Engineering Quality Review Team (EQRT) consisting of representatives from Design Engineering as well as customer and oversight organizations has been established at IP3.

The responsibilities of the Engineering Assurance Program and the EQRT involve oversight of all areas of design engineering including the process of maintaining and controlling the plant's design bases. Engineering Assurance and the EQRT conduct assessments, participate in design control audits and assign "expert teams" to evaluate and strengthen specific design and design control processes.

Engineering weaknesses identified by any of the above means provide the basis for corrective actions and further trending to ensure their effective resolution.

The Engineering Assurance Program is designed to: (1) ensure that engineering and design control weaknesses are promptly identified through self assessments, customer feedback and trending, (2) initiate and oversee action to eliminate identified weaknesses and (3) monitor the effectiveness of completed corrective action.

Several engineering process improvement items were implemented as a result of EQRT initiatives. Such items included design process improvements to minimize avoidable Engineering Change Notices (ECNs), post modification testing, calculations, design control and procedural improvements.

NEI 96-05 Assessment

The NEI 96-05 project was a self assessment of a sample of the Indian Point 3 licensing bases and the programs for maintaining them. It was conducted to identify missing or incorrectly applied programmatic elements that could (or did) lead to licensing bases discrepancies.

The NEI 96-05 Project followed the NEI initiative directions (Draft NEI 96-05 "Guideline for Assessing Programs for Maintaining the Licensing Bases"). The project sampled two safety and two non-safety related systems. A total of 36 FSAR discrepancies were identified, of which none were identified as having safety significance, and 2 of the findings were identified as having potential regulatory significance. Actions were identified to ensure that these discrepancies are resolved.

Management Observations

The Management Observation Program is designed to promote management awareness of in-plant conditions and performance. The program is focused primarily on observable facets of personnel performance, both in operations and in maintenance. Although it is not intended to function as an element of design configuration control, the program has, on occasion, identified design issues needing resolution. Individual issues identified through the Management Observer Program are tracked either by PID, DER or ACTS or by the Minor Deficiency List.

The Management Observation Program tasks senior and middle managers to conduct periodic, "on scene" assessments of key plant programs and processes. The evolution is monitored, and comments, if any, are immediately shared with participants. Any safety issues are immediately addressed through the normal process.

The Management Observation Program emphasizes personal performance with particular attention being given to training effectiveness, work practices, safety rule compliance and procedural adherence. The goal is to ensure that senior managers at the plant have an accurate perception of how people execute their duties in routine, in-plant evolutions. In addition, managers gain an excellent perspective on the effectiveness and efficiency of the processes that control the progress of work and watch standing in the plant.

About 4,000 management observations were conducted during 1996. A substantial number of these observations involved Design Engineering, System Engineering, walkdowns and testing activities.

Post Modification Testing Process Assessment

In 1994, to improve performance in post modification testing (PMT), the Authority initiated a review of this process at IP3. A task force consisting of NYPA employees and consultants with extensive experience in nuclear engineering, construction, operation and pre-operational and startup testing was established in January 1995 to review the post modification testing process.

Testing associated with 512 modifications, minor modifications and design changes prepared by Site Engineering, Design Engineering and Technical Services were reviewed. The review addressed the adequacy of test requirements and acceptance criteria specified in the modification package, the incorporation of these test requirements and acceptance criteria in the test procedure, and the performance of 308 completed tests.

A review package consisting of the modification package, test procedure and completed test, if testing had been performed, was obtained. If the completed test was not available, only the modification package and test procedures were reviewed.

The following conclusions were drawn from the post modification testing review:

- Approximately 14% of the test procedures contained a testing deficiency.
- The majority of the deficiencies had little or no significance to plant safety. Only one modification was found to contain a deficiency of a nature that leads to a less than adequate level of confidence that safety function would be met. Re-testing of this modification demonstrated that the design and installation were correct and that the safety function would be accomplished.
- Test performance, documentation and resolution of exceptions was found to be a strength.
- The rate of deficiencies did not appear to vary significantly among engineering departments or disciplines.

As a result of this review, the following actions were taken or initiated:

- Testing standards utilized for post maintenance testing were provided to the post modification test group for use when appropriate.
- Training was provided to engineers responsible for preparation of test procedures on the types of deficiencies found.
- Engineering assurance personnel review the quality of a sample of new test procedures and provide feedback to the engineering department.
- Enhanced procedural guidance on post modification testing requirements was developed and implemented.

Post Maintenance Testing Assessment

To determine if similar deficiencies existed in post maintenance testing, a review of a sample of post maintenance tests completed during the 1994 Restart Outage was performed. The results of this review indicated that the performance of post maintenance testing was satisfactory .

Preventive Maintenance Program Improvements

The Preventive Maintenance (PM) program was enhanced during the RS 94 outage. The recent PM enhancements at IP3 included a review of procedure adequacy, scheduling and confirmation that appropriate technical information was properly reviewed for inclusion into the program. Components exhibiting a higher than industry average failure rate at IP3 were analyzed for PM program adequacy.

A database assessment of the PM program was conducted. This involved review of PM program findings from NRC, INPO and QA. Plant personnel were interviewed and other plants with effective PM programs was contacted. The goal of this review was to determine whether there were any commitments and/or potential commitments relating to the IP3 component preventive maintenance program should be incorporated.

As a result of this assessment, over 1,000 documents identified in numerous databases were reviewed. The IP3 Licensing and ORG (SORs, IEINs, INTs, SERs, etc.) databases were searched for issues dealing with Preventive Maintenance. The purpose of these reviews was to identify commitments that were made which concerned the IP3 PM program. When a commitment was identified, existing IP3 programs and/or documents were reviewed to verify that the commitment was implemented.

INPO ASSESSMENTS AND NRC INSPECTIONS

INPO Assessments

The Institute of Nuclear Power Operations (INPO), conducts assessments and evaluations of site activities to make any assessment of plant safety, to evaluate management systems and controls and to identify areas needing improvements.

During the September 1996 evaluation, the INPO team examined station organization and administration, operations, maintenance, engineering support, training and qualification, radiological protection, chemistry, and operating experience review. The team also observed the actual performance of selected evolutions including surveillance testing.

INPO identifies beneficial practices and accomplishments as well as areas in need of improvement. Significant areas in need of improvement were identified during the September INPO Evaluation include: (1) work management process and large backlog of corrective maintenance and overdue preventive maintenance activities; (2) delays in resolving equipment problems because of insufficient control of engineering work; (3) alignment of engineering and maintenance to common priority and goals; (4) management for equipment deficiencies and (6) training weaknesses. The Authority is presently reviewing these deficiencies. INPO findings and recommendations are used by the Authority to improve all aspects of its nuclear program.

NRC Inspections

Certain NRC inspections performed at IP3 also provide confirmation of the consistency of SSC configuration and performance with the design bases. Findings resulting from these inspections provide opportunities for improvements. Such inspections include the following:

Electrical Distribution System Functional Inspection (EDSFI)

The Electrical Distribution System Functional Inspection (EDSFI) was performed by the NRC in April 1991. Seventeen EDSFI issues were identified during this inspection. Such issues included loading of EDS busses, voltage calculations, design control, fuse control, EDG transient and static loading, AC and DC systems coordination, etc. All 17 EDSFI issues were evaluated and resolved.

Several programmatic improvements mentioned in the electrical area above were a result of the EDSFI inspection. Other EDSFI related improvements include the following:

- Operating Procedures (EOPs, SOP-EL-15) were revised to incorporate operator load management for maintaining 480 Volt System loading within analyzed limits (loading of EDS busses).
- Maintenance activities around 480V switchgear are administratively restricted during monthly EDG testing to reduce the probability of a 3 phase fault at 480V switchgear (AC Fault Analysis).
- The Fuse Control Program was implemented and required procedures to maintain fuse control have been established.

- EDS Design Bases Calculations were revised or reconstituted to demonstrate the adequacy of the existing configuration.

- Seven of the seventeen EDSFI issues required FSAR revisions.

In follow-up inspections, the NRC reviewed and verified the closure of all EDSFI issues.

Special NRC Team Inspection

From April 26 through May 28, 1993, the NRC performed a special team inspection at IP3. There were three principal objectives of this inspection were (1) to assess IP3's identification and evaluation of root causes for the declining performance, (2) to assess how effectively IP3 identifies and corrects problems, and (3) to identify restart issues other than those already identified that must be corrected before restart of IP3.

The NRC team determined that the root causes for the declining performance of IP3 were weak managerial processes, controls and skills. Specifically, the team found that the corrective action process was ineffective in identifying and resolving problems; that deficiencies existed in the commitment tracking system that resulted in several missed commitments for the fire protection/Appendix R program; that many plant organizations had weak and missing procedures; that backlogs in the areas of maintenance, engineering, technical services, quality assurance and operating experience program had not been adequately controlled and evaluated for safety significance; that the emergency diesel generator had not received the required preventative maintenance because vendor recommendations had not been incorporated into the program; that several major changes in the organization and changes in major programs had not been managed well; that the planning, scheduling and work control process was ineffective; that communications among organizations and within organizations were poor; and that there was a general lack of accountability.

As a result of the above finding, the Authority undertook several corrective actions including the following:

- Established a site wide Commitment Tracking System. An improved system is expected to go on line early 1997.

- Updated the Appendix R and Fire Hazards Analysis (FHA) to reflect current issues.

- Development of a Fire Protection Design Basis Document.

- Conducted an inspection and implemented improvements in fire seals and barriers throughout the plant.

- Re-organized and relocated the Design Engineering Division and clarified responsibilities for design basis.

Service Water System Operational Performance Inspection (SWSOPI):

An inspection of the Service Water System was performed by the NRC in 1993 and a follow up inspection was performed in 1994. The inspection was performed to assess the adequacy of NYPs compliance actions with NRC Generic Letter 80-13, Service Water System Problems Affecting Safety Related Equipment. The basis of the inspection was for the NRC to verify that the Service Water System was designed, operated, tested, and maintained such that its intended safety functions would be provided, if needed, thus ensuring the public health and safety during postulated design basis events. The inspection did not identify any thermal hydraulic performance issues that would adversely affect the systems capability to perform its designed safety function. Approximately eighteen issues regarding system corrosion, design control, and maintenance were identified. Of the eighteen only one issue was identified against procedures requiring a change to an System Operational Performance, which was closed in December 1994. The remaining items have been individually addressed by NYPA and closed with exception of one (1) item, a Service Water Pump Selector Switch modification. This issue is open pending modification by Electrical Design Engineering with a scheduled completion by the next refueling outage (Spring 1997).

NRC Readiness Assessment Team Inspection (RATI)

The Readiness Assessment Team Inspection (RATI) was conducted at IP3 from April 3-21, 1995. The RATI performed an independent, broad scope assessment of management controls, programs and personnel to support a safe restart and the continued operation of IP3. The RATI evaluated the areas of Management Programs, Independent Oversight, Self-Assessment, Operations, Maintenance and Surveillance and Engineering and Technical support.

The team identified six issues, in addition to the issues identified by IP3 staff, that required action prior to plant restart. Of the six NRC identified restart issues, five were focused on the quality of plant documentation regarding the quality of plant procedures, drawings and posted information to assist plant operators, (Operator Aids). Specifically, the six NRC identified issues involved the following areas:

- Alarm response procedures not referencing the alarm actuating device or alarm setpoints.
- The Auxiliary Feedwater Pump Building temperature controllers were not set in accordance with the system drawings and the temperature controllers and fans were not routinely functionally tested.
- The load schedules located inside of certain electrical distribution panels were not controlled documents and did not match the system drawings. The load schedules posted inside the panels did not reflect plant modifications that had added or removed loads.
- The closeout process for setpoint changes was not clearly proceduralized. The setpoint change control procedure and process did not ensure that all procedures and documents affected by a setpoint change were revised.
- 122 Document Change Requests (DCR) were backlogged against the "Type A" (control room vital) drawings. The team concluded that the information provided in the DCRs should be available to the operators prior to plant restart.
- The team found that a design change turnover had been completed by the responsible engineer without the adequate review or concurrence by the Operations Department as required by plant administrative procedures. The NRC concluded that a review of similar design change closeout packages should be conducted prior to the completion of the inspection.

The NRC confirmed that each of the issues had been or would be adequately addressed prior to restart.

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

As a result of the Performance Improvement Plan (PIP), and the Restart and Continuous Improvement Plan (RCIP), the process and methods for identification, tracking, evaluation and resolution of plant problems were revised and improved. Some of the improvements include independent evaluations being performed on current equipment status and operating parameters, as well as the safety screening process for procedure changes being strengthened. Management has re-emphasized the need for an aggressive questioning attitude.

Nuclear Administrative Policy, NuAP 1.1, "Nuclear Safety", states that all personnel shall apply a questioning attitude towards facility operations with the objective of improving safety and performance. It also requires personnel to identify conditions adverse to quality and propose corrective and preventive action necessary to enhance the safe and efficient operation of the plant. Plant procedures require that identified conditions adverse to quality be evaluated to determine their impact on operability and reportability and be assigned a priority for timely corrective action based on their significance to safety. The primary means of capturing these conditions are the Problem Identification Description (PID) process and the Deviation & Event Reporting and Operability Determination Procedure (DER) process. Other means include Operability Reviews, Industry Operating Experience (OE) Review, Design Document Open Item (DDOI), Request for Document Change (RDC), Speakout and QA audits.

Problem Identification Description (PID) Process

The PID process is part of the work control process and is used to report deficiencies in plant material conditions. The PID, as described in Station Directive SPO-SD-01, is used to identify a known or suspected problem or deficiency with a plant system, structure or component (SSC), or used to request modification or engineering services. This work control process has several levels of operability/ reportability reviews built into it. The initial preparer of the PID considers operability/reportability concerns. The PID is subsequently reviewed by a licensed Senior Reactor Operator (SRO) in Operations. An additional review is performed by the PID subcommittee and/or the Work Control Group on each normal working day. If as a result of these reviews, an operability/reportability concern is raised, then this equipment problem is entered into the Work Control system, as well as the Deviation Event Report (DER) system to capture the issue for plant management review for further evaluation.

Deviation & Event Reporting (DER) Process

For identification of problems and implementation of corrective actions, the DER process was adopted in 1993 to replace the Significant Occurrence Reports (SOR). This new process captures the significant events as well as adverse quality conditions and the lower threshold events that could be of value to the plant in understanding and correcting the lessons learned from operating experience. This process is controlled through Administrative Procedure AP-8.

As each DER is initiated, the issue is assessed for potential operability and reportability by the initiator and/or plant supervision. DERs determined to be potentially reportable or that potentially affect system or component operability are automatically and personally forwarded to the shift manager for formal operability and reportability determination.

The Operations Review Group (ORG) preliminarily screens each new DER on each normal working day and assigns a significance level, A through D, based on how the DER affects nuclear safety, plant operation, and protection of the public health and safety. The significance level is used to determine the safety importance of the item so that the appropriate category (1, 2, or 3) level of analysis is assigned. ORG screening also includes checking operability / reportability determination assignments and assigning responsibility for DER evaluations. The evaluating department is normally selected based on expertise, function and knowledge of the issue and causes. A DER Review Committee reviews DERs on each normal working day and concurs with, or recommends revision of, screening and evaluation assignments based on knowledge of the issue.

Significance level A DERs involving human performance are presented to the Performance Enhancement Review Committee (PERC). Significance level B and C DERs are selected for PERC presentation by the Plant Leadership Team (PLT). The PERC consists of members of the PLT and from the ORG, QA or Training Departments. The objective of the PERC is to gain a timely understanding of the human performance cause of the event and review lessons learned or identify corrective actions necessary to prevent recurrence. Prompt review of these issues allows for wide, rapid determination of corrective actions when necessary (sometimes implemented through plant standdowns).

A DER evaluation may be a response, Industry Operating Experience (OE) evaluation, investigative critique, equipment failure evaluation, root cause analysis, post transient review, or some combination of these evaluation types. The above evaluations are required by AP-8.2 or 21.2 to address the cause, the extent of condition, and proposed actions to prevent recurrence. The IP3 plant management has established an expectation that all evaluations be completed within 21 days of identification. For the DERs tracked by ORG, the current average evaluation completion time is about 28 days and the long term trend is decreasing. This evaluation response time is tracked in the ORG monthly DER self assessment report. Starting in January 1997 the DERs, tracked by both ORG and QA, will have the evaluation completion time tracked. DER evaluations are reviewed and approved by senior site management. Significance level A (highest significance level) root cause analyses are reviewed by the PLT and the Plant Operating Review Committee (PORC), and significance level B root cause analyses are reviewed by the PLT.

The DER system is used for program, process and human performance monitoring as well as for equipment monitoring and trending. In order to ensure that appropriate evaluations and corrective actions are identified and implemented under the IP3's 10CFR 50.65 Maintenance Rule (MR) program, a DER is initiated in accordance with AP-62 whenever a system is categorized as (a) (1).

Through the DER evaluation process, necessary corrective actions, or proposed actions to prevent recurrence are identified, reviewed and approved by senior site management. These completed DER evaluations are then returned to the ORG or QA for final processing. Level A and B DERs are reviewed by ORG or QA to assure that the corrective actions proposed should prevent recurrence and that completion of the proposed actions are tracked in a formal tracking system. Level C DERs are selectively reviewed in the same manner by ORG or QA.

The DER process provides guidance for identification of problems, issues and deficiencies at a low threshold. Any plant personnel can initiate a DER for any occurrence or issue. The DER process has been used by the IP3 staff for identifying potential problems with the FSAR, design basis and Technical Specifications. There were approximately 50 DERs written, between

November 1993 and December 1996, that shows these types of concerns are identified in the DER process. As previously described, the DER screening and analysis program would assure that these identified concerns receive the appropriate level of analysis and operability/reportability reviews.

Since November of 1993, there have been a number of enhancements to the corrective action program based on plant personnel feedback and operating performance. ORG continues to assess the program based on inspection audits, assessments, observation of daily performance, and through contact with other utilities within the industry.

In May of 1995 the NRC issued Inspection Report 95-80 for the Readiness Assessment Team Inspection (RATI). The report concluded that an effective management team, corrective action program and oversight function were in place to support a safe plant restart. The report also stated that the deviation/event report process and the corrective action program were sufficiently established to identify and resolve plant deficiencies in a timely manner. In January of 1996, the NRC issued the results of a special inspection (Report 95-16) which included a performance based team inspection to assess IP3's ability to identify and correct problems. The conclusion was that well-defined programs are in place that allow for the identification, tracking and closure of problems. It also noted QA's ability to identify problems and adverse trends and to propose appropriate corrective actions. On the negative side, the report noted that negative trends were evident in the timeliness and numbers of backlogs for several departments. It was also concluded that the quality of deficiency resolution was mixed.

A conclusion drawn from the various QA audits and NRC inspection reports completed in 1995 and 1996 is that the Authority identifies problems well, but has mixed results in performing analyses and completing timely corrective actions. Audits conducted by QA in 1995 and 1996 and Assessment and Trend reports jointly prepared by ORG and QA also concluded that the implementation of the timely corrective action program was weak and, in some cases, did not prevent problems from recurring. To resolve this issue, the level A and B evaluations and corrective actions are now approved by the appropriate General Manager and presented to the PLT for approval. Also extensions for corrective action items require approval from the Plant Manager (level A) or General Manager/Director (level B). The weekly performance indicator report is reviewed by senior management with emphasis on reducing overdue ACTS items, reducing DER evaluations greater than 40 days old and reducing DER preventive corrective actions greater than six months old.

Any adverse conditions identified by the Audit Program are documented on DERs completed in accordance with AP-8 requirements. Any corrective action and follow-up activities are tracked by QA. QA reviews the evaluation, corrective actions and final DER closure.

Overall, approximately 7400 DERs have been processed since the start of the DER process in 1993. Out of that total, approximately 89% have been identified by workers, supervisors and managers, 7% by QA/QC and 4% by the NRC and other outside agencies.

Operability Reviews

Operability reviews are conducted in accordance with AP-8 in order to determine the operability of structures, systems, or components (SSCs) that have been identified as being in a degraded or nonconforming condition. Operability determinations are required to be made with a timeliness commensurate with the potential safety significance of the issue. Operability concerns are typically identified by PIDs, DERs, engineering design reviews, Design Basis Document reviews, walkdowns or testing. Operability determinations are resolved in accordance with AP - 8. The Shift Manager (SM) is required to ensure that operability determinations are sufficient to address SSC capability to perform its safety related function.

Industry Operating Experience (OE) Program

To facilitate IP3 exposure to Operating Experience issues, ORG screens the INPO Nuclear Network to identify issues of IP3 interest. Operating Experience Issues that raise an operability or reportability issue are documented with a DER. Operating Experience issues that warrant corrective actions or documented reviews are entered in the IP3 ACTS process. An executive summary is prepared by ORG of Operational Experience issues of interest to IP3 that are identified in the INPO Nuclear Network and is routed to key site managers.

The IP3 ACTS process provides the mechanisms to track, schedule and assign action items and document reviews. NRC Restart Issue 41 corrected the backlog of OE issues and reviewed the adequacy and effectiveness of previous OE issues. The work for this restart issue eliminated much of IP3s excessive backlog of OE items, and established the ACTS process as the mechanism for assigning, prioritizing, scheduling, trending, managing and reporting OE issues.

Informational routings of all Operating Experience documents from the INPO Nuclear Network are available on the plant computer under the NYPA bulletin board. These routings are transmitted from JAF to IP3 normally on each working day. Approximately 15 days of the most current information is available on the NYPA bulletin board.

The Operating Experience program generates approximately 300 ACTS items for resolution of OE type issues each year.

Currently ORG discusses current industry operating experience with both licensed and non-licensed operators during requalification training.

Design Document Open Item (DDOI) Process

The DDOI is used to document and track missing and/or discrepant information in the Design Basis Documents (DBD). The ACTS database is used to track DDOIs to closure. Current procedure CMM 2.1 requires that DERs be written for Priority I and discrepant Priority II DDOIs.

Request for Document Change (RDC) Process

The Request for Document Change (RDC) is used in accordance with Administrative Procedure AP 18.8. This document was previously discussed in secion (a) of this letter. During the evaluation of a change request, a DER may be initiated to report the potential for an abnormal plant condition, a design issue, a potentially reportable condition or a potential safety issue.

Speakout

Nuclear Administrative Policy NuAP 1.9, "Employee and Contractor Concerns and Protection," establishes the methods for employees and contractors to express nuclear safety or quality concerns and describes the protection afforded to employees and contractors who raise such concerns or provide such information to their supervisor, Authority management, the Nuclear Regulatory Commission (NRC) or any other Federal or State agency.

Employees and contractors are encouraged to initially raise any nuclear safety or quality concerns to their supervisors or Authority management for resolution and to use the established Authority procedures for reporting conditions adverse to quality. This policy established the Nuclear Safety SPEAKOUT Program to provide employees and contractors with a further opportunity to report nuclear safety or quality related concerns in a manner that provides protection for the worker and assures that the concerns are properly addressed. It is the policy of the Authority that its employees and contractors fully cooperate in any investigations and be candid and forthright when interviewed or asked to provide information in connection with an investigation.

In accordance with the Nuclear Safety Speakout Program procedure (NSS-1), the Speakout Evaluation Committee consists of the Speakout Manager, Speakout Administrator, a plant General Manager, the QA Manager and a legal representative. This committee gives assurance that issues identified to Speakout get the proper management attention.

QA Audits

The Quality Assurance Audit Program is established to verify that required program activities are performed in accordance with NYPAs Quality Assurance Program. Specifically, Criterion XVIII of 10CFR50, Appendix B applies to audit activities, with details of audit activities specifically described in Quality Assurance Procedure, QAP-18.1.

The objectives of the QA Audit Program are to determine that the QA Program, and supporting procedures, have been developed, documented and maintained in accordance with licensing, regulatory, and other commitments, and that the program has been effectively implemented, as verified by the examination and evaluation of objective evidence of conformance.

An Audit Schedule is developed in order to verify that Technical Specification requirements are met. This function is delegated to the QA organization by the Safety Review Committee (SRC), via procedure SRCP-9. These responsibilities include the quality criteria contained in 10CFR50, Appendix B. Other audits may be performed to assure the adequacy of, and conformance with, the overall program requirements.

Audit activities affecting design bases and configuration, including major program audits, such as Design Control, Fire Protection/Appendix R, Security, etc., are conducted by the QA organization, with technical support, as required. Audits are conducted in other areas, as required, to support plant operation, maintenance, testing and engineering activities.

The audit activities described above are established for each audit and address the following:

- Ensure that applicable operating, maintenance and test procedures covered by the audit are addressed, by review of selected procedures and verification of affected activities.

- Ensure that applicable FSAR sections, Technical Specifications, Operational Specifications and other regulatory requirements are addressed in audit activities.

- Ensure that specific commitments made to the NRC affecting the audit activity are suitably addressed by review and verification of activities affecting the commitments.

- Ensure that a review of audit subject and suitable references is made to the FSAR and overall effect on FSAR content. Any deviations between FSAR and site processes, SSCs or configuration are flagged, documented, as required, by DERs and corrective actions are pursued.

A review of audits performed over the past three years confirms that these issues are actually addressed in the audits for operating, maintenance and testing procedures, and that suitable reference is provided in the audits, as applicable.

Rationale for concluding that system, structure and component configuration and performance is generally described in the individual audits. A review of audits performed over the past three years confirms that these issues are actually addressed in the audits, and that suitable reference is provided in the audits, as applicable. The specific audits conducted which affect design bases activities for system, structure and component configuration and performance do verify that suitable rationale is provided in the design activities performed to ensure that this requirement is met.

Any discrepancies or adverse conditions identified by audits are documented on a DER, completed in accordance with AP-8 requirements. Prior to implementation of the DER system, any such conditions were documented on Corrective Action Requests (CARs), included with the completed audit. Corrective action is performed by the responsible department and any followup activity is performed by the QA organization or ORG, as needed, to resolve the item and attempt to avoid recurrence. Any required follow-up activity resulting from audit activities, including Audit Recommendations, are documented in the ACTS and tracked by individually assigned ACTS items. QA and ORG also perform trending of DERs. The organization responding to the DER is also responsible to evaluate extent of condition, including problem follow-up and to avoid recurrence.

A review of DERs for audits affecting design activities over the past three years revealed that numerous DERs were issued for specifics of the design aspect of the audited activity. None of these DERs which were issued were classified as significant.

ACTIONS TO PREVENT RECURRENCE

Root Cause Analysis

The Deviation and Event Analysis program, described in Administrative Procedure AP-8.2, divides all DERs into four "significance" levels, A, B, C and D, with "A" being the most significant and "D" being the least significant. The responsibility for performing evaluations is placed with the department that has the most responsibility and experience for the area of concern. Level D DERs are entered into the database for "trending" purposes. The other three levels, A, B and C are assigned to responsible departments for analysis.

Once a DER is designated a significance level, a Root Cause Analysis category can then be assigned for the evaluation of the DER. There are three levels of root cause analysis that are performed at IP-3, category 1, category 2, and category 3 with category 1 being the most intensive and category 3 the least. The categories of analysis are generally synchronized with the significance level assigned to a DER, e.g., a DER significant screened as a level A will normally be assigned a category 1 and a DER screened significance level B will normally be assigned a category 2 analysis and so on. Because of the investigative attributes that have been incorporated within the root cause analysis program, the FSAR, Design Basis and Technical Specification issues are dealt with in a formal manner.

The IP-3 program for root cause analysis is specifically delineated in the DER Analysis Manual. This manual was generated in September of 1995 to provide a standard upon which the IP-3 staff could perform analysis of plant events. The program was based on the principles of the Institute for Nuclear Power Operations (INPO) Human Performance Program. IP-3 reviewed other utility programs and incorporated the best principles of those programs also.

Some staff members at IP3 are trained in various root cause analysis techniques. Non-equipment related level A and B DERs are performed by individuals trained in observation techniques and root cause analysis. Level A/B/C DERs related to equipment failures are performed by individuals specifically trained in equipment failure analysis. These qualifications are tracked by the training department via a root cause matrix.

Unlike significance level C and D, significance level A and B DERs are given an acceptance examination by ORG or QA. The DER evaluation results are also required to be presented to senior management, PLT and/or PORC prior to submittal to ORG for closure.

Trending & Self - Assessment

The trending program has been developed as a sub-section of the DER process. Departmental procedure ORG-AD-004, Rev 1 (effective 4/30/96), In-house Event Trending, contains the direction provided for the performance of in-house trending. The trending process helps to ensure that the plant is operated and maintained within the design basis by identifying events or trends that indicate that corrective actions were not effective in preventing repeat events. On a monthly basis (procedure ORG-AD-004 identifies quarterly) ORG prepares an assessment report that is presented to PORC and provided to Department Managers to use in assessing their department's performance and areas/opportunities for enhancement within their department. Based on the assessment, specific recommendations or requests for evaluations are assigned and captured in ACTS. The status of previous trend report recommendations or adverse trends are updated in subsequent reports until the issue is resolved.

When a sufficient number of similar events have occurred, a DER is written to document the unacceptable number of occurrences. A DER would also be written for an adverse trend. This DER would be given a higher significance level (normally a level B category) than the individual events and therefore a higher level evaluation would be required (usually an investigative critique, category 2 evaluation). This higher level of evaluation increases the depth of analysis of the combined events. The number of higher significance (level A or B) DERs has averaged about six to seven per cent of the total number of DERs for 1996. Trend DERs, unacceptable level of occurrence DERs, repeat events, maintenance rule identified systems and significant individual events are included within this grouping.

A quarterly assessment report is issued using the ORG monthly reports, QA monthly reports, QA audits, NRC inspection reports, LERs and INPO reports.

The area of identification of repeat events has been noted as requiring improved methodology and is being addressed by a business plan action item (ACTS#19951). This ACTS is to review and implement recommendations and is due to be completed March 31, 1997.

ACTION & COMMITMENT TRACKING SYSTEM (ACTS)

In November of 1993, ACTS was introduced at IP3. The software was then made available in February 1994 to site personnel at the personal computer level on the site network. Prior to the introduction of this system, commitments for the plant were tracked on individual local systems that did not allow for routine site wide access.

The ACTS is a computerized database management system used at IP3 to administer the corrective actions program. All corrective actions resulting from significance level A DERs and significance level B DERs are required to be tracked through final implementation by ACTS.

The ACTS can also be used to track corrective actions initiated from level CDER evaluations or level D DERs which have uncompleted corrective actions. However, it is also permissible to track these actions through other auditable tracking systems available to plant staff such as Action Plans (APLs), Problem Identification or Work Request (PID or WR), Engineering Change Notice (ECN), or Request for Document Change (RDC). In this usage, an auditable tracking system means a system that is administratively controlled by an approved written procedure.

The system is widely and heavily used. Since November 1993 there have been approximately 24,000 ACTS entered. The use of ACTS has established control over plant commitments and corrective actions that has worked for the benefit of the plant. Because of the high visibility of the ACTS system many people have the ability to review and be familiar with what is being proposed at a site wide level. Overdue ACTS items are tracked in the management Weekly Performance Indicators report.

The ACTS system is controlled by Administrative Procedure AP-37.4, Action & Commitment Tracking System. Some of the ACTS items are given special control such that acceptance of their completion must be approved by department managers and ORG or QA and their schedule more closely monitored. Therefore, important issues such as dealing with design basis would be entered into a controlled ACTS item to ensure completion/resolution.

REPORTING TO THE NRC

The processes for identifying and reporting problems to the NRC are the reportability review and Emergency Notification System (ENS) notification portion of the DER process under AP-8 and the reportability requirements of AP-8.1 for Licensee Event Reports (LERs).

NYPAs method to comply with 10 CFR 21 is proceduralized in AP-8, AP-8.2 and NLP-7. These procedures require that any employee having information reasonably indicating the existence of a defect or failure to comply shall promptly report this condition using the DER process. NLP-7 provides guidance on how to determine if the information available reasonably indicates a potential defect or failure to comply. Once a DER has been initiated, ORG assigns the Part 21 evaluation to the appropriate department for resolution. This evaluation determines if a defect could create a substantial safety hazard or if a non-compliance is associated with a substantial safety hazard, and thus whether or not the condition is reportable to the NRC under 10CFR21. If the evaluation determines that a substantial safety hazard does exist, NLP-7 details the steps necessary to report this information to the NRC. In addition, NLP-7 discusses the time frame requirements for evaluation and NRC report ability to ensure that issues are resolved within the time allowed by 10CFR 21.

(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 Authority recognizes that assuring that the design bases information is adequate and available is a critical element in plant operation. The processes for controlling design information have been evolving and improving since Indian Point 3 became operational; further improvements are under way or are in the planning stages.

Overall, the Authority's current processes and programs are effective in assuring that the configuration of the Indian Point 3 Nuclear Power Plant is consistent with their design bases because: (1) the current processes and programs provide reasonable assurance that changes to the plant and its operating, maintenance and testing procedures are reflected in the design bases; (2) Design Basis Document programs provide a baseline for engineering information and reasonable assurance that the engineering information is accurate, complete and readily available; (3) audits and inspections conducted by the Authority, the NRC and other industry organizations identify when weakness or discrepancies exit and they are captured and corrected by the Corrective Action Program; (4) and, that current process to identify problems and implement corrective actions are designed to determine the extent of the problem, prevent the recurrence of problems and assure that reports to the NRC are submitted in accordance with the requirements of 10 CFR 50.

Design and Configuration Control Processes

The Authority's procedure hierarchy is made up of Nuclear Administrative Policies (NuAPs), Administrative Policies (APs), and technical procedures. NuAPs are divided into eight categories: (1) Nuclear Policy, (2) Organization and Responsibilities; (3) Engineering Functional Requirements; (4) Regulatory Functional Requirements; (5) Operations and Maintenance Requirements; (6) Administrative Functional Requirements; (7) Training; and (8) Security. NuAPs establish roles and responsibilities for overall responsibilities, document control, and the deviation and event analysis programs. Plant-specific APs detail procedures for compliance with NuAPs. Technical procedures prescribe requirements for the operation, maintenance, and testing of structures, systems and components. Technical procedures are controlled by APs. Specific technical procedures implement the requirements of 10 CFR 50.59 and 50.71(e). Appendix B requirements are implemented by the Indian Point 3 Quality Assurance Program and other technical procedures.

Operating, Maintenance and Testing Procedures

Procedural controls have been established that require changes to procedures and new procedures to be compared to the design bases contained in the FSAR and approved prior to use. When a discrepancy between a procedure and the existing plant configuration is identified, the discrepancy is resolved and appropriate changes to either the plant or the procedure are made.

Engineering and configuration control processes require that changes to design bases information be assessed for potential effects on procedures. Required procedure changes are identified and tracked until completion.

Audits and self-assessments support the conclusion that in general, design bases requirements are translated into operating, maintenance and testing requirements. Audits are done of operating and maintenance procedures related to the implementation of Technical Specification requirements. NRC inspections have also reviewed the operating, maintenance and testing procedures.

Structure, System and Component Performance

Walkdowns, testing and configuration programs provide information on the condition of Indian Point 3 structures, systems and components. Walkdowns compare the as-built configuration of the plant with plant records. Configuration programs outline processes to provide additional assurance that plant records are updated and complete. Test programs confirm that the performance of the structure, system or components is consistent with the design bases.

Problem Identification and Corrective Actions

The Deviation Event Report (DER) and other problem identification/corrective action programs provide for prompt identification, documentation and correction of conditions adverse to quality, including those which could have a significant effect on quality or nuclear safety. Corrective actions are based on an appropriate level of causal analysis and provide steps to prevent recurrence.

The DER problem identification process is closely linked with the reportability process. Problems are screened as they are identified for reportability. Approximately 7400 DERs have been processed since the start of the process in 1993.

Performance Improvements

During an approximate 2 1/2 year shutdown for performance improvements, the Authority performed reviews of its processes and programs as part of its Restart and Continuous Improvement Plan. Significant improvements were made to support the decision to restart the plant in 1995.

The Authority has multiple self assessment processes which includes QA, Independent Safety Engineering Group, and an Engineering Quality Review Team that evaluates the Design Control function. These processes identify both strengths and weaknesses. Deficiencies are placed in the plant wide corrective action program and are evaluated for operability, reportability, extent of condition and significance. Problems are prioritized and worked off in accordance with their significance to plant safety.

As a result of our self assessment process, and other inspections performed by the NRC and INPO, some weaknesses were identified in the area of setpoint control, material conditions, engineering backlogs, procedure adequacy and effectiveness of corrective action. All areas are being addressed through our corrective action program.

While acknowledging the need to improve, the Authority remains confident in it's ability to safely operate Indian Point 3. Audits and inspections also recognized strengths and recent accomplishments. Improved operations, instrument and control, and maintenance line management involvement in training and fewer human performance events were cited as beneficial practices and accomplishments by INPO. A QA Design Control Audit completed in June 1996 concluded that the design control process was being adequately implemented although there were some deficiencies identified. Our most recent 4th Quarter Integrated Self-Assessment/Trend Report noted improvements in work backlog reductions and the corrective action program.

Using trending techniques, causal analysis and extent of condition reviews, the corrective action program provides a vehicle for continuous improvement. Additionally, through a review of Deviation/Event Reports (DERs), which are a main part of our corrective action process, it is apparent that the Indian Point 3 staff has developed a questioning attitude and are finding design control issues and resolving them.

Further information supporting the basis for these conclusions is detailed in sections (a) through (d) and (f) of this report.

(f) Design basis document (DBD) program

The Design Basis (DBD) Program for the New York Power Authority began in the spring of 1987. The intent of the DBD Program was to consolidate widely scattered design basis information and to identify missing and discrepant information within the design documentation of the plant. The Authority notified the NRC of its intent to embark on a pilot design basis consolidation program in its response to the SSOMI findings in the summer of 1987.

The Authority developed a pilot program for the initial development of the DBDs for the Main Steam, RHR and Auxiliary Feedwater Systems between 1987 and 1989. Based on the results of the pilot program the development of seventeen additional system and topical DBDs using the original A/E and NSSS vendor as a team was authorized. The Authority later authorized the preparation of two additional DBDs, Fire Protection and High Energy Line Breaks Outside Containment, for a total of twenty-two (22) DBDs for, covering about forty (40) systems and three (3) topical areas. Preparation of additional DBDs such as Containment Spray and Emergency Diesel and Appendix R Diesel Generators are being planned.

The IP3 DBDs cover the majority of the safety related systems. The Authority's DBDs are based on a Format and Writer's Guide which was developed as part of the pilot program. The Writer's Guide was enhanced from lessons learned from the pilot program, a review of DBDs from other utilities, and recommendations by the original designers of IP3.

The format of the Authority's Writer's Guide dictates that all DBDs clearly identify each requirement, its origins by reference and a logical reason for the requirement. In addition, the format specifies that for each requirement a design feature with references be listed. This provides assurance that the requirement is in fact implemented in the plant design.

The DBD content includes the following:

- design background and scope of responsibility of the A/E and NSSS vendor in the design of each system
- system and component level requirements
- regulatory and licensing requirements
- system interfaces, interlocks and actuation features
- accident analysis assumptions
- a complete list of references used in the DBD
- a listing and summary of calculations highlighting the purpose, assumptions and results
- a listing and summary of modifications to trace the changes since the original design

Design Document Open Item (DDOIs) were generated due to discrepancies between design documents, discrepancies between design documents and the FSAR, and when source documentation for design requirements or design features was missing. The team of United Engineers and Constructors (now Raytheon Engineers and Constructors) and Westinghouse was informed at the start of the contract that the Authority was to be informed immediately as soon as any safety significant issues were discovered.

The Authority, in its review of DDOIs during and after the completion of the DBDs, followed the guidelines of Authority procedures CMM-2.2 and CMM-2.1 to deter the priority of DDOIs with respect to operability and reportability. The DDOIs were prioritized based on their potential safety significance. DDOIs involving discrepant information which could adversely impact the operation of a sturture, safety related system, or component were classified as Priority I. DDOIs involving missing or discrepant information which supported the design basis of a safety related system, structure, or component, but which did not directly impact operability, were classified as Priority II. DDOIs involving missing or discrepant information which supported the design basis of a system, structure, or component (safety related or non-safety related) were classified as Priority III. DDOIs involving missing or discrepant information which merely supplemented the design basis of a system, structure, or component (safety related or non-safety related) were classified as Priority IV. Open Priority I and II DDOIs were evaluated for potential operability concerns prior to plant restart in 1995.

Recently, CMM-2.2 was incorporated into CMM-2.1, and now CMM-2.1 controls the program for DBDs and DDOIs, including a link to 1) the ACTS database for tracking DDOIs to closure and 2) the DER process for documenting issues requiring operability/reportability reviews. The Authority has performed a review of all Priority III and IV DDOIs which are classified as missing for possible deferral since open items in this category are of low safety significance and may not warrant as much attention as the higher priority DDOIs. Deferral was conservatively assumed to mean that the DDOI resolution is inactive and would be closed only if the resolution activity coincided with day to day work of the Authority where extra resources were not required. The candidates for deferral were evaluated against four criteria: 1) the Appendix R safe shutdown equipment list, 2) the PRA, 3) the SQUG safe shutdown components required following a seismic event, and 4) the components required to minimize onsite or offsite radiological dose consequences. Thus, if an Open Item Priority III or IV DDOI involved only missing information on a system, structure or component and was not included in the above components, it was deferred. However, the item was still retained in the DBD, and if an ongoing modification or analyses during the normal workload could address the issue, the DDOI could still be resolved as part of that project.

As part of the contract for development of DBDs, the original designers searched their archives to collect design documentation which was used as a reference in the DBD and transmitted it to the Authority as part of the deliverables. Documentation received was entered into the Authority's Nuclear Document Control System. Any proprietary information was identified and retained by the NSSS vendor. Calculation summaries were provided by the NSSS vendor for proprietary calculations.

The accuracy of the DBDs was enhanced by 1) having the DBDs reviewed and accepted by the Authority per the DCM-11 process, 2) the DBDs were prepared in accordance with an approved Appendix B Quality Assurance Program, 3) Verification of correct transfer of source documents was built into the program through a cross-review conducted by the original designers prior to submitting a draft to the Authority.

Procedure CMM-2.1, provides guidance on preparation, review, approval and updating of Design Basis Documents. The update process is performed by the issuance of a Pending Change Notice (PCN) to the DBD. This is a form that documents any design changes such as a modification, a new or revised calculation, or open item resolution that needs to be incorporated into the DBD. The PCN, once approved, is issued by Document Control to the controlled copy holders of the DBDs. The copy holders are instructed to maintain the PCNs in the front of the DBDs until a formal revision is issued incorporating the information. All currently issued PCNs and draft PCNs still in review will be formally incorporated into revised DBDs in the current ongoing DBD Update Project.

As a result of weaknesses identified in maintaining DBD's up-to-date the Authority is undertaking a DBD update project. The DBD update project is being undertaken with the assistance of an outside vendor to research all pertinent information since the original issuance of the DBDs and update the DBDs to incorporate all outstanding PCNs as well as other information requiring incorporation. The order in which the DBDs are being revised is based on the PRA risk significant systems. Included in this effort is a review of all open and deferred DDOIs to determine if any information has been generated that may resolve a DDOI. The objective is to resolve all Priority I and II DDOIs. As part of the DBD update project a review of the FSAR is also being performed to ensure that all the information in the DBDs and the FSAR is consistent. The effort is currently ongoing with a scheduled completion date in the third quarter of 1997.

The Authority has begun a DBD validation process. This process is designed to provide reasonable assurance that the design basis information contained in the DBDs is reflected in the as-built plant and in those documents used to operate, maintain and test the plant systems. This validation is a vertical slice of selected design basis requirements and attributes. The specific methodology and other details are given in the Authority's procedure NEAP-38.

To date, the validation of the AFW system has been completed. This review did not identify any operability concerns although there were several design descrepancies identified. The open issues uncovered during the validation of the AFW system program for resolution were entered into the ACTS.

The Authority is currently undertaking the validation of the 125 Volt DC Electrical Distribution System and other DBD system validations will be undertaken in a scheduled manner.

In addition to the DBD Consolidation Program, the Authority has performed reconstitution in certain areas as new calculations or revisions to calculations were warranted (for modifications). As a result of analyses performed for special projects (such as 480Volt, 125Volt, DC, HVAC) new calculations were developed. These are discussed further in the response to request (c).

List of DBD Systems and Topics

DBD No.	TITLE/SYSTEMS
DBD-301	Main Steam
DBD-302	Residual Heat Removal
DBD-303	Auxiliary Feedwater
DBD-304	Service Water
DBD-305	Instrument Air
DBD-306	Safety Injection -Low Head Safety Injection
DBD-307	480V AC, 125V DC, 120V Vital AC
DBD-308	Component Cooling Water/Spent Fuel Pit Cooling
DBD-309	Nuclear Instrumentation - Excore - Incore - Neutron Flux
DBD-310	Seismic Piping and Supports
DBD-311	Chemical Volume and Control
DBD-312	Reactor Protection/Engineered Safeguards
DBD-313	Rod Control
DBD-314	Reactor Coolant
DBD-315	Heating Ventilation & Air Conditioning - Central Control Room - Primary Auxiliary Building - Fan House - Control Building - Emergency Diesel Generator Building - Electrical Tunnel - Containment - Fuel Storage Building - Shield Wall Area - Auxiliary Feedwater Building
DBD-316	Containment Integrity Systems - Isolation Valve Seal Water - Hot Penetration Cooling - Hydrogen Recombiners - Weld Channel Containment Penetration Pressurization
DBD-317	Feedwater

DBD No.	**TITLE/SYSTEMS**
DBD-318	Seismic Buildings and Structures
DBD-319	Condensate/Condensate Polishing
DBD-320	Radiation and Environmental Monitoring - Area Radiation Monitoring - Process Radiation Monitoring - Post Accident Sampling
DBD-321	Fire Protection - Water Supply and Distribution System - Fixed Fire Suppression Systems - Portable Fire Suppression Equipment - Fire Detection and Alarm System - Fire Hazards Analysis - Fire Resistive Building Features - Smoke Removal Systems - Plant Drains - Safe Shutdown Analysis - Appendix R Emergency Lighting System - Appendix R Communication System
DBD-322	High Energy Line Break Outside Containment

LIST OF ACRONYMS

AABD - Accident Analysis Basis Document
AC - Alternating Current
ACRS - Advisory Committee on Reactor Safeguards
ACTS - Action Commitment Tracking System
ADM-SD - Administation Station Directive
AFW - Auxiliary Feedwater
ANSI - American National Standards Institute
AP - Administrative Procedures
ARP - Alarm Response Procedure
ASME - American Society of Mechanical Engineers
CAR - Corrective Action Requests
CCWS - Component Cooling Water System
CGI - Commercial Grade Items
CIM-AD - Configuration Information Management Administative Directive
CMM - Configuration Management Manual
CNO - Chief Nuclear Officer
DBD - Design Bases Document
DC - Direct Current
DCM - Design Control Manual
DCR - Design Change Request
DCR - Document Change Resolution
DDOI - Design Document Open Item
DDS - Design Drafting Standards
DE - Design Engineer
DEE-SD - Design Engineering Electrical Station Directives
DER - Deviation and Event Reporting
DP - Differential Pressure
ECCF - Electrical Change Control Form
ECN - Engineering Change Notice
ECRIS - Electrical Cable and Raceway Information System
EDG - Emergency Diesel Generator
EDS - Electrical Distribution System
EDSFI - Electrical Distribution System Functional Inspection
EDSM - Electrical Distribution System Model
EES - Electrical Engineering Standard
ENS - Emergency Notification System
EOP - Emergency Operating Procedures
EPRI - Electrical Power Reasearch Institute
EQ - Environmental Qualification
EQRT - Engineering Quality Review Team
ERG - Emergency Response Guidelines
ESM - Engineering Standards Manual
ESP - Earth Surface Potential
FHA - Fire Hazards Analysis
FIN - Fix-It-Now Team
FPES - Fire Protection Engineering Standards
FPP - Fire Protection Program
FSAR - Final Safety Analysis Report
GL - Generic Letter

HS	-	Hand Switch
HVAC	-	Heating Ventilation and Air Conditioning
ISEG	-	Independent Safety Engineering Group
I&C	-	Instrument & Control
ICD	-	Interface Control Document
IEEE	-	Institute of Electrical and Electronics Engineers
IEIN	-	NRC Information Notice
IES	-	Instrumentation and Control Standards
INDMS	-	Nuclear Data Management System
INPO	-	Institute of Nuclear Power Operations
INT	-	IP3
IP3	-	Indian Point 3 Nuclear Power Plant
IPE	-	Individual Plant Examination
ISD	-	Instructional System Design
ISI	-	Inservice Inspection
IST	-	Inservice Testing Program
JAF	-	James A. FitzPatrick Nuclear Plant Power
JCO	-	Justification for Continued Operation
LCOs	-	Limiting Conditions of Operations
LER	-	Licensee Event Report
LDE	-	Lead Design Engineer
LSFT	-	Logic System Functional Test
MCM	-	Modification Control Manual
MOV	-	Motor Operated Valve
MPFFS	-	Maintenance Preventable Function Failure
MR	-	Maintenance Rule
MRL	-	Modification Responsibilities List
NEAP	-	Nuclear Engineering Administrative Procedure
NEI	-	Nuclear Energy Institute
NFPA	-	National Fire Protection Association
NLP	-	Nuclear Licensing Procedure
NRC	-	Nuclear Regulatory Commission
NSAC	-	Nuclear Safety Analysis Center
NSE	-	Nuclear Safety Evaluation
NSEIS	-	Nuclear Safety and Environmental Impact Screens
NSS	-	Nuclear Safety Speakout
NSSS	-	Nuclear Steam Supply System
NuAP	-	Nuclear Administrative Policies
NUMARC	-	Nuclear Management and Resource Council
NYPA	-	New York Power Authority
OD	-	Operation Directive
OE	-	Industry Operating Experience
ONOP	-	Off Normal Operating Procedures
ORG	-	Operations Review Group
OWA	-	Operator Work Around
PC	-	Pressure Controller
PCN	-	Pending Change Notice
PEDB	-	Plant Equipment Data Base
PEP	-	Preliminary Engineering Package
PERC	-	Performane Enhancement Review Committee
PIPS	-	Problem Identification Description

PIP	-	Peformance Improvement
PLT	-	Plant Leadership Team
PM	-	Preventive Maintenance
PM	-	Plant Manager
PMT	-	Post Modification Testing
POP	-	Plant Operating Procedures
PORC	-	Plant Operations Review Committee
PRA	-	Probablisistic Risk Assessment
PTO	-	Protective Tagging Order
PWM	-	Procedure Writers Manual
QA	-	Quality Assurance
QC	-	Quality Control
QSR	-	Quality Safety Reviewers
QTR	-	Qualified Technical Reviewer
RATI	-	Readiness Assessment Team Inspection
RAS	-	Reasonable Assurance of Safety
RCIP	-	Restart and Continuous Improvement Plan
RDC	-	Request for Document Change
RE	-	Responsible Engineer
RHRS	-	Residual Heat Removal System
RMT	-	Restart Management Team
RPO	-	Responsible Procedure Owner
SAR	-	Safety Analysis Report
SAT	-	Situation Assessment Team
SCR	-	Setpoint Change Request
SED	-	Site Engineering Department
SED-AD	-	Site Engineering Department Administation Directive
SER	-	Safety Evaluation Report
SFPCS	-	Spent Fuel Pool Cooling System
SM	-	Shift Manager
SOP	-	System Operating Procedure
SOR	-	System of Records
SQUG	-	Seismic Qualification Utilities Group
SRC	-	Safety Review Committee
SRCP	-	Safety Review Committee Procedure
SRO	-	Senior Reactor Operator
SSC	-	Structures Systems and Components
SSOMI	-	Safety System Outage Modification Inspection
STRIP	-	Surveillance Test Results Improvement Program
STP	-	Surveillance Testing Program
SWS	-	Service Water System
TPC	-	Term Procedure Change
TS	-	Technical Specifications
TSP	-	Tube Support Plate
UE&C	-	United Engineers and Constructors, Inc.
UFSAR	-	Updated Final Safety Analysis Report
USQ	-	Unreviewed Safety Question
VPE-PC	-	Vice President Engineering and Project Controls

VETIP	-	Vendor Equipment Technical Information Program
WACP	-	Work Activity Control Procedure
WCC	-	Work Control Center
WPO	-	White Plains Office
WR	-	Work Request