Stephen E. Quinn
Vice President

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Indian Point Station
Broadway & Bleakley Avenue
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Telephone (914) 734-5340

February 13, 1997

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

SUBJECT: Response to Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information, NRC letter from James M. Taylor to Eugene McGrath dated October 9, 1996

In accordance with the subject request in the NRC letter of October 9, 1996, received by Eugene R. McGrath on October 16, 1996, Consolidated Edison Company of New York, Inc. ("Con Edison") is providing in the attachments hereto the information requested by the Commission.

Con Edison, in preparing its response, has consolidated its current commitments in the nature of voluntary initiatives to enhance Indian Point Unit No. 2 (IP 2) design bases processes in Attachment A hereto. All commitments of this response are set forth in Attachment A. Attachment B, in the form of a report, contains the information sought by the Commission.

The response to information request (a) describes our engineering design and configuration control processes, including preparation of safety evaluations, updates of the Updated Final Safety Analysis Report (UFSAR), and application of quality assurance requirements.

The response to information request (b) provides our rationale for concluding that the design bases requirements documented as a result of the processes described in the response to (a) above are reflected in operating, testing, and maintenance procedures.

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In responding to information request (c), we base our rationale that system, structure and component configuration and performance are consistent with design bases on the information pertaining to Con Edison processes, procedures, programs, and projects described in the responses to (a) and (b) above, the response to information request (d), and the additional information request concerning design reviews. Information is also provided on the many assessments, test programs, and operations support programs that contribute to the reasonable assurance of consistency between the design bases and plant systems, structures, and components.

The information set forth in our response to request (d) describes our processes for identifying design and configuration problems of in-plant or industry origin, for determining the extent of their applicability at IP 2 and reportability to the NRC, and for taking actions to correct problems and prevent their recurrence.

Historically Con Edison has implemented review and upgrade programs intended to improve the capability of IP 2 systems, structures, and components to perform design functions. In the course of implementing such upgrade programs some design bases reconstitution has occurred. Resulting modifications were accomplished through the processes and programs described in Attachment B, providing further support for the conclusion that the plant's configuration is consistent with the design bases.

Based upon the foregoing, and in response to request (e), Con Edison has reasonable assurance that: its current processes and programs are sufficient to maintain the plant configuration consistent with the design bases; design bases requirements are properly translated into design specifications and operating, maintenance and testing procedures; the configuration of structures, systems and components are consistent with design bases; and that deviations are reconciled as they are identified.

We trust that the information set forth in the attachments is responsive to your request. Should you or your staff have any questions concerning this submittal, please contact either the undersigned or Mr. Charles W. Jackson, Manager, Nuclear Safety & Licensing.

Very truly yours,

Attachments

cc: Mr. Samuel Collins, Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Mr. Hubert J. Miller
Regional Administrator - Region I
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Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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| In the Matter of | ) | |
| | ) | |
| CONSOLIDATED EDISON COMPANY | ) | Docket No. 50-247 |
| OF NEW YORK, INC. | ) | |
| (Indian Point Station, | ) | |
| Unit No. 2) | ) | |

Mr. Stephen E. Quinn, being duly sworn, states that he is Vice President, Nuclear Power, Consolidated Edison Company of New York, Inc.; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this response to the Request for Information Pursuant to 10 CFR 50.54(f); that he is familiar with the content thereof; that he has overseen the development of the Company's response, which has included information received from the nuclear steam supply system vendor and other contractor personnel; that he has directed verification of the response through the Company's Quality Assurance organization; and that based on the processes employed in preparing the response and the reviews performed, the information presented is correct to the best of his knowledge, information and belief.

BY: Stephen E. Quinn

Stephen E. Quinn
Vice President

Subscribed and sworn to
before me this 13th day
of February, 1997.

Karen L. Lancaster
Notary Public

KAREN L. LANCASTER
Notary Public, State of New York
No. 60-4643659
Qualified In Westchester County
Term Expires 9/30/97

50-247 CEC INDIAN POINT 2

RESPONSE TO NRC RAI RE ADEQUACY AND AVAILABILITY OF DESIGN BASES INFO.

REC'D W/LTR DTD 02/13/97....9702190330




# - NOTICE -

ATTACHMENT A

COMMITMENTS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY, 1997

## Commitments

Discussions and descriptions of processes contained in responses to the NRC's information requests (a) through (e) are not intended to establish new commitments. The following are those commitments and/or areas of enhanced management attention which Con Edison currently intends to voluntarily initiate and complete consistent with the provisions of the October 9, 1996 Taylor letter (page 6 at fn.8):

- UFSAR review program scheduled for completion within the next 24 months.

  This program will include the following elements:

  1. Verification of the accuracy of UFSAR design basis information.
  2. Review to confirm that the UFSAR design basis information is properly reflected in plant operation, maintenance, and test procedures.
  3. Review the UFSAR to identify and resolve any internal disagreements or inconsistencies which could impact the design basis.
  4. Development of a process to enhance overall UFSAR accessibility.

- Continuation of the Design Basis Document (DBD) Initiative

  1. Supplementation of the currently existing 22 DBDs with a combination of additional DBDs and added information on interfacing systems to existing DBDs within the next 24 months.
  2. Verification of the compatibility of the design basis requirements in the UFSAR with new and existing DBDs.
  3. Development of a process to enhance overall accessability and retrievability of DBD information, and keep DBD information current.

- Continuation of the SSFA program with at least one SSFA to be performed per year.

- Development and implementation of a training module to increase awareness of the significance of design basis information, its documentation and maintenance. This training will be completed by December 31, 1997 for personnel currently assigned.

ATTACHMENT B

RESPONSE TO REQUEST FOR INFORMATION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FEBRUARY, 1997

TABLE OF CONTENTS

ACRONYMS & ABBREVIATIONS

| | |
|---|---|
| AC | Alternating Current |
| AD | Administrative Directive |
| ALARA | As Low As Reasonably Achievable |
| AOV | Air Operated Valve |
| ASME | American Society of Mechanical Engineers |
| CAMP | Corrective Action Monitoring Program |
| CCR | Central Control Room |
| CI | Corporate Instruction |
| CITRS | Condition Identification and Tracking System |
| COL | Checkoff List |
| CRDM | Control Rod Drive Motor |
| DBD | Design Basis Document |
| DBI | Design Basis Initiative |
| DC | Direct Current |
| DMCS | Drawing Management Control System |
| DMD | Design Modification Drawing |
| DMRG | Daily Management Review Group |
| DOE | Determination of Equivalency |
| EDG | Emergency Diesel Generator |
| EDS | Electrical Distribution System |
| EDSFI | Electrical Distribution System Functional Inspection |
| EOP | Emergency Operating Procedures |
| EPRI | Electric Power Research Institute |
| ERG | Emergency Response Guidelines |
| ESMS | Electrical System Management Software |
| EQ | Environmental Qualification |
| ESP | Engineering Support Personnel |
| FP | Fire Protection |
| FSAR | Final Safety Analysis Report |
| GIP | Generic Implementation Procedure |
| GL | Generic Letter |

| | |
|---|---|
| HVAC | Heating, Ventilating and Air Conditioning |
| IEEE | Institute of Electrical and Electronics Engineers |
| IP 2 | Indian Point Unit No. 2 |
| IPE | Individual Plant Examination |
| IPEEE | Individual Plant Examination for External Events |
| ISI | Inservice Inspection |
| IST | Inservice Testing |
| keff | Measure of Core Reactivity ("k-effective") |
| kV | Kilovolt |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| LP | Low Pressure (version of ERG) |
| MCC | Motor Control Center |
| MET | Meteorological |
| MOV | Motor-Operated Valve |
| MPFF | Maintenance Preventable Functional Failure |
| NCTS | Nuclear Commitment Tracking System |
| NFSC | Nuclear Facilities Safety Committee |
| NP | Nuclear Power |
| NPE | Nuclear Power Engineering |
| NQA | Nuclear Quality Assurance |
| NRC | Nuclear Regulatory Commission |
| NUMARC | Nuclear Management and Resources Council, Inc. |
| NUPOP | Nuclear Parameters and Operations Package |
| OEM | Original Equipment Manufacturer |
| OIR | Open Item Report |
| OSTI | Operational Safety Team Inspection |
| PC | Periodic Calibration |
| PI | Periodic Inspection |
| PFRT | Project File Review Team |
| PMA | Project Managing Authority |
| PPMIS | Power Plant Maintenance Information System |
| PRA | Probabilistic Risk Assessment |

| | |
|---|---|
| PSA | Probabilistic Safety Assessment |
| PT | Periodic Test |
| PWR | Pressurized Water Reactor |
| QAPD | Quality Assurance Program Description |
| RES | Request for Engineering Services |
| RFO | Refueling Outage |
| ROI | Report of Installation |
| ROR | Radiological Occurrence Report |
| RSAC | Reload Safety Analysis Check List |
| RSE | Reload Safety Evaluation |
| SAO | Station Administrative Order |
| SAS | Safety Assessment System |
| SE | Safety Evaluation |
| SIL | Safety Impact Level |
| SIQ | Safety Impact Questionnaire |
| SNSC | Station Nuclear Safety Committee |
| SOR | Significant Occurrence Report |
| SQUG | Seismic Qualification Utility Group |
| SSC | Systems, Structures, and Components |
| SSFA | Safety System Functional Assessment |
| SSFI | Safety System Functional Inspection |
| SWS | Senior Watch Supervisor |
| SWSOPI | Service Water System Operational Performance Inspection |
| TNMS | Tag Number Management System |
| TS | Technical Specifications |
| UFSAR | Updated Final Safety Analysis Report |
| USQ | Unreviewed Safety Question |
| WOG | Westinghouse Owners Group |

## DEFINITIONS

Design Bases:

As identified in 10 CFR 50.2, design bases are defined as "that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or range of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted 'state of the art' practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals". The design bases, as so defined, are a subset of the licensing basis and are contained in the UFSAR. Information developed to implement the design bases is contained in other documents, some of which are docketed and some of which are retained by the licensee.

Design Document:

A document belonging to the set of documents comprised of design input documents, design studies or analyses, and design output documents that specify the design of a structure, system, or component. These are documents to which one can refer to verify that structures, systems and components have been designed to perform their intended function within the reference bounds of the controlling parameters and that form the point of departure for future plant modifications. (NUREG-1397)

Class A:

The determination of which systems, structures and components affect safety is in accordance with 10 CFR 50 Appendix B and includes those:

- which comprise or are necessary to insure the integrity of the reactor coolant pressure boundary;
- which ensure the capability to shutdown the reactor and maintain it in a safe shutdown condition;
- whose failure could result in conservatively calculated offsite doses that exceed 0.5 Rem to the whole body or its equivalent to any part of the body; and

- whose failure (structures only) could reduce the functioning of plant features within the above categories to an unacceptable safety level.

Those structures, systems, and components of the plant that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public are designated Class A.

Class 1E:

The safety classification of the electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment.

Contractors:

Outside vendors contracted for various expert assistance (i.e., Burns & Roe, ALCO Diesel, Ebasco, Conax, Foxboro, Stone & Webster, United Engineers and Constructors (UE&C), Westinghouse Electric Corporation, etc.)

## 1.0 Summary and Conclusions

Since the initial licensing of Indian Point Unit No. 2, Consolidated Edison Company of New York, Inc. (Con Edison) has employed programs, processes and procedures that, collectively, provide the Company with reasonable assurance that plant configuration is controlled and that operations, maintenance and testing is conducted in a manner consistent with the design bases.

Activities that contribute to this assurance include plant upgrades, initiatives, and Nuclear Quality Assurance Department assessments such as Safety System Functional Assessments (SSFAs). Upgrades to the design documents and to the processes for preparing and installing plant modifications have added to the depth of design bases knowledge.

The Con Edison Quality Assurance program is described in a Corporate Instruction, CI-240-1, Quality Assurance Program for Operating Nuclear Plants. A section of this procedure describes Nuclear Power Engineering's responsibilities for the use of appropriate design criteria, applicable regulatory requirements, and the design bases as it relates to plant modification. These and other provisions are detailed in OP-290-1 Section 5, Engineering Procedures for Operating Nuclear Power Plants, which contain the engineering design and configuration control process for modifications to the IP 2 plant. OP-290-1 is one of the processes that implements requirements of 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B of 10 CFR 50.

To date, programs have been completed to improve the engineering design and configuration control process in areas such as procedure and drawing updates and modification package reviews. On-going programs, which supplement the engineering and configuration control process, include drawing management control and nuclear safety classification. Programs which address additional NRC requirements, such as Environmental Qualification and the Fire Protection Program, are reviewed for applicable requirements in the engineering design and control process.

The original FSAR was finalized in 1970. Its first update following issuance of 10 CFR 50.71(e) took place in 1982 and it was thereafter referred to as the UFSAR. During the process, design documents were compiled from site sources, vendors, and the original architect-engineer to facilitate access to design information. The UFSAR has been updated under 10 CFR 50.71(e). Critical UFSAR Chapter 14 Safety Analyses updates for each fuel reload have been performed by the NSSS vendor and checked by plant staff.

Con Edison, in the fall of 1996, commenced informational review of the IP 2 UFSAR for correctness and consistency with plant-controlled procedures.

As a result of the preliminary FSAR review we initiated in the later part of 1996 and our efforts in preparing the response to this information required, we are expanding the scope of our FSAR review program as described in Attachment A, Commitments.

Con Edison has reasonable assurance that design bases requirements are translated into operations, maintenance and test procedures for the following reasons and as set forth in detail in Section 3.2 below. First, IP 2 operating, maintenance, and testing Procedures are governed and controlled by Station Administrative Orders (SAOs) and departmental administrative procedures. Second, in the early 1980s an FSAR review was performed in conjunction with a review of licensing commitments. This resulted in an update of the operating procedures to be consistent with the FSAR. In addition, the Westinghouse Owners Group developed Emergency Response Guidelines that became the bases for plant-specific Emergency Operating Procedures that were simulator validated. The implementation of this integrated set of procedures enhanced station work processes and better enabled plant personnel to control design, operations, maintenance, and testing activities consistent with the design bases. Additionally, procedures for updating the design bases in a consistent manner when plant systems are modified are provided in a Station Administrative Order. Moreover, results of many reviews and audits conducted of operations, maintenance and test activities confirm Con Edison's belief that there is reasonable assurance for concluding that the IP 2 design bases have been appropriately translated into operations, maintenance and testing procedures.

The extensive use of procedures, multiple internal and external assessments, evaluations, audits and inspections, successful operations and testing programs, and many improvement and upgrade programs provide reasonable assurance that the IP 2 systems, structures and components (SSCs), configuration and performance are consistent with the design bases. This rationale is outlined immediately below and more fully described in Section 3.3 of this report.

Multiple levels of management processes provide reasonable assurance IP 2 SSCs are consistent with the design bases. These processes include control of engineering, operations, maintenance, and test processes, including both the use of procedures and training in the use of procedures, which are intended to assure that the design bases are properly considered. Assessments, both internal and external, provide evidence that the processes are properly conducted and that the IP 2 SSCs are consistent with the design bases. Identified discrepancies are added to the plant corrective action programs. Processes that support operations such as operability determinations, walkdowns and testing programs are intended to provide additional assurance that the IP 2 SSCs are consistent with the design bases. Additional specific initiatives and programs enhance and contribute to the accuracy of the information that comprises the design bases and the consistency of the SSCs with the design bases by reviewing and upgrading existing design information or generating new information as required. The results of these implemented processes and programs provide reasonable assurance that IP 2 is operated and maintained within the design bases and that deviations are managed by the plant corrective action systems.

Corrective action processes provide reasonable assurance that deficiencies are identified and corrected. These processes are discussed in Section 3.4 of the report. Active employee identification of conditions potentially adverse to quality include procedural discrepancies, and equipment and documentation deficiencies. The identified problems are recorded, evaluated, tracked, and dispositioned by the corrective action processes. IP 2 has recently instituted the use of the Condition Identification and Tracking System (CITRS). CITRS tracks identified deficiencies or events, assignment of actions and action due dates, status update of actions, and contains or references results of evaluations and investigations.

Regular assessments of the status of corrective actions are performed by station departments, management, and the Nuclear Quality Assurance Department. Trending reports are published to enhance management's ability to assess the potential impact of trends and to assist in the determination of the effectiveness of the corrective actions to prevent event recurrence. Management reviews of the corrective action program and problem identification processes are also performed. Management is continuing to stress the importance of responding to corrective actions more promptly and of increasing the effectiveness of problem analyses. Although seldomly occurring, issues identified relating to design bases are entered into appropriate corrective actions system for resolution. Should any similar design basis issues be identified in the future, they will be handled in the same appropriate manner.

IP 2 has implemented various programs, plant upgrades, design changes, and system assessments. Although Con Edison does not have a formally designated reconstitution program, the IP 2 design bases have been and continue to be reconstituted on a case-by-case basis as needed. Section 4.0 of the report describes a number of examples of major programs where design bases information has been upgraded and reconstituted, where appropriate. In addition, a number of major programs, projects, and upgrades, characterized as review programs, are also described.

The collective weight of the programs and processes described in the responses to the information request enables Con Edison to provide reasonable assurance that the configuration of IP 2 is effectively controlled and is consistent with the design bases.

## 2.0 Introduction

This response provides information requested by the NRC letter, "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information," James M. Taylor to Eugene R. McGrath, dated October 9, 1996. Descriptions of programmatic activities and initiatives found in this response are not commitments. Those specific commitments made by Con Edison are listed in Attachment A, Commitments.

Con Edison, as the sole owner and operator of Indian Point Unit No. 2 (IP 2), understands the importance of configuring, operating, maintaining and testing the plant in accordance with its design bases. The term "design bases" is defined as set forth in 10 CFR 50.2. The term "design document" is defined as set forth in NUREG-1397.

Operation of the plant in conformance with its design bases is key to assuring public health and safety. Con Edison has a longstanding and continuing policy of maintaining compliance with the design bases. This is reflected in appropriate Con Edison policies, procedures and practices.

IP 2 is a Westinghouse 4-loop pressurized water reactor located on the eastern shore of the Hudson River in Buchanan, New York, which is approximately 35 miles north of New York City. The plant was licensed for full-power operation in September, 1973. Regulatory requirements and nuclear industry capabilities have changed over the years. Con Edison has expended significant resources over the life of the plant to maintain reasonable assurance that IP 2 is operated in conformance with its design bases. During the operating life of IP 2, documentation regarding plant upgrades, improvement initiatives, design reviews and audits have supplemented the plant's design information which is compiled in design documentation. Independent assessments and inspections [(e.g. Safety System Functional Inspections (SSFIs) and Safety System Functional Assessments (SSFAs)] serve an audit function and identify needs for further supplemental design information. Additional assurance of the adequacy of the plant configuration control processes is provided by other assessments, walkdowns, test programs, and a variety of programs described in Sections 3 and 4. Collectively, these activities have provided additional assurance for consistency among design bases information, plant configuration, and procedures. Processes are in place to identify and resolve any identified discrepancies among design information, the as-built structures, systems and components (SSCs), the Updated Final Safety Analysis Report (UFSAR), and related documentation.

This response provides a summary description of the policies, procedures and programs to provide an overview of efforts to maintain the plant configuration consistent with the design bases. It describes processes currently in-place which are subject to change. Improvement and clarification of existing procedures and programs is ongoing.

The following sections address items (a) through (e) and the request concerning design review and design reconstitution programs in NRC's October 9, 1996 letter. The information provided describes elements of the processes that provide reasonable assurance that IP 2 continues to be operated in accordance with its design bases and the rationale for concluding that the processes that control the plant configuration are effectively implemented.

## 3.0 Information Requested by the NRC

Sections 3 and 4 of this report provide the Consolidated Edison Company of New York, Inc., response to information requested in the NRC letter dated October 9, 1996 from Mr. James M. Taylor to Mr. Eugene R. McGrath, Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information.

### 3.1 Request

**(a) Description of Engineering Design and Configuration Control Processes, Including Those That Implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR Part 50.**

Response

Con Edison processes and programs serving to maintain the design bases are grouped and described in Sections 3.1.1 through 3.1.4 as the current engineering design and configuration control process, current programs, historical programs, and safety evaluation and analysis. A summary follows.

Con Edison's Quality Assurance Program describes Nuclear Power Engineering's responsibilities for the use of appropriate design criteria, applicable regulatory requirements, and the design bases as defined in 10 CFR 50.2 relating to plant modifications. The engineering design and configuration control process for the Indian Point Unit No. 2 (IP 2) plant is contained in OP-290-1, Section 5.0, which includes Sections 5.1 through 5.25. It serves primarily to provide a control process for plant modifications. OP-290-1 Section 5 is one of the processes that implements requirements of 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B of 10 CFR Part 50.

Programs that supplement the engineering design and configuration control process are the Drawing Management Control System, the Engineering Assurance Program, and the Nuclear Safety Classification Process. The Drawing Management Control System serves to control the engineering design process for drawings and to support the primary design procedure for modifications. It ensures that the contents of the master drawings are proper and supports design bases information in the as-built process.

The Engineering Assurance Program provides a continuous process of self-assessment to attain a goal of excellence in engineering and design work; objectives include responsibility and accountability within the engineering organization and affirm engineering quality and excellence through a continuous process of self-assessment.

The Nuclear Safety Classification Process recognizes that not every portion of each of the listed systems, structures, and components contributes to preventing or mitigating the consequences of postulated accidents. Therefore, an individual component, part, or commodity may be evaluated to determine the proper nuclear safety classification for engineering and safety evaluations.

Environmental Qualification, Seismic Qualification, Fire Protection and the American Society of Mechanical Engineers (ASME) Section XI are current programs that address NRC requirements, which are reviewed in the engineering design and configuration control processes.

In response to Generic Letter (GL) 87-02 and GL 87-02, Supplement 1, the Seismic Qualification Utility Group (SQUG) was formed and developed the "Generic Implementation Procedure for Seismic Verification of Nuclear Plant Equipment."

The Fire Protection Program Plan (FPPP) provides a consolidated description of the IP 2 Fire Protection Program, incorporating the various commitments and descriptive information in the program.

IP 2 was designed prior to the development and implementation of ASME XI. Therefore, ASME XI requirements were not part of the original plant design bases. With the implementation of ASME XI, IP 2 Inservice Inspection (ISI) Programs, including relief requests, were developed and submitted to the NRC. Today, the IP 2 ISI programs have the practical effect of adopting ASME XI as a design basis except where relief requests identify exceptions to the ASME Code.

Historically, programs have been initiated to improve the engineering design and configuration control process. Modification packages began to use a new Design Modification Drawing process in 1985. The process improved the clarity of both record design drawings and drawings identifying pending plant modifications. Engineering procedures governing modification processes were upgraded in 1989 to provide improvements to the engineering process. Resulting improvements enhanced clarity and more fully addressed the completeness and consistency of documentation, interfaces, and reviews for modification packages. Information was added and made more retrievable.

From 1988-90, a review of the IP 2 modification packages and files was conducted. Approximately 1,300 modifications and files were reviewed to determine completeness. A retrieval effort was undertaken to collect the information identified as missing from the individual modification files.

The IP 2 procedure for compliance with 10 CFR 50.59 delineates the requirements for preparing and documenting Safety Evaluations for design modifications (including equivalent replacements, setpoint changes, temporary repairs and jumpers) and for certain new procedures and changes to procedures. Guidance from the NRC and industry documents is in the procedure.

Safety Evaluations are performed for many IP 2 processes and in particular are implemented for modifications through the OP-290-1 engineering process. The 10 CFR 50.59 evaluations are used to identify updates to the Updated Final Safety Analysis Report (UFSAR). Updates to the UFSAR include the effects of changes made to the facility or procedures described in the UFSAR, Safety Evaluations performed in support of requested license amendments or conclusions that changes have not involved an unreviewed safety question (USQ) (10 CFR 50.59 process).

Since 1973, Westinghouse has provided the analysis for the core designs and fuel fabrication for Con Edison. Con Edison maintains the UFSAR and makes necessary changes to the UFSAR and plant procedures as required for each fuel cycle.

### 3.1.1 Engineering Design and Configuration Control Process

#### 3.1.1.1 Quality Assurance Program

Con Edison's Quality Assurance Program is described in a corporate instruction, CI-240-1, Quality Assurance Program for Operating Nuclear Plants. Section II of the corporate instruction contains provisions regarding design control. It stipulates that Nuclear Power Engineering is responsible for the use of appropriate design criteria, adherence to applicable regulatory requirements, and conformity with the design basis as defined in 10 CFR 50.2 as it relates to plant modifications. Other provisions of the corporate instruction stipulate that independent reviews of design documents be accomplished to assure that appropriate quality standards are specified. These and other provisions are detailed in a Nuclear Power Engineering implementing procedure OP-290-1 Section 5, Engineering Procedures for Operating Nuclear Power Plants.

#### 3.1.1.2 Modification Process Description

The engineering design and configuration control process for the IP 2 plant primarily provides a control process for plant modifications. Con Edison's Engineering Operation Manual, OP-290-1, Section 5.0, which includes section 5.1 through 5.25, describes this process. OP-290-1 is one of the processes that implements requirements of 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B of 10 CFR Part 50. The following paragraphs briefly describe the OP-290-1 procedures and the configuration control process embedded in those procedures. Figure 3.1, Engineering Modification Process, shows the eight elements of the plant modification process.

# Engineering Modification Process

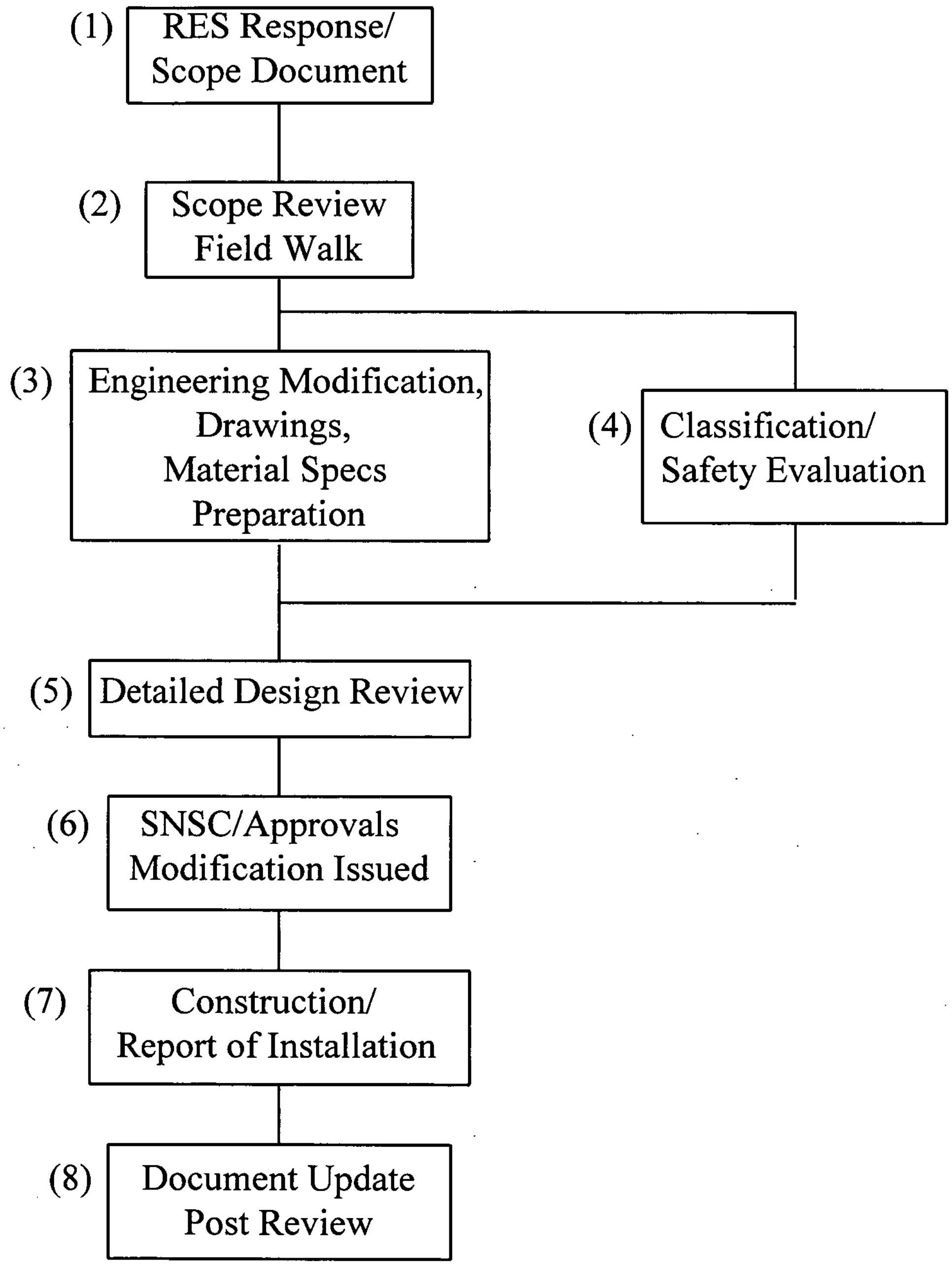


Figure 3.1

**Element 1** begins the process with a request for modification and development of the project scope, described in SAO-415, Requests for Engineering Services. Under normal conditions, the user generates a modification request through a Request for Engineering Services (RES). This request initiates an engineering evaluation and recommendation. If a plant modification is required to resolve the user's request, a project is established. A project prioritization process, which looks at 11 key factors including heavily weighted consideration of potential impact on core melt frequency and accident mitigation, ranks the goals and urgency of specific projects among the candidates. This system is biased so that projects affecting nuclear safety receive appropriate attention.

The preparation of the Routine Modification Package begins with on-site Nuclear Power (NP) concurrence on a project scoping document that defines parameters and goals. This is the first of many joint reviews by both on and off-site NP organizations, which are contained in a single business unit reporting to the same vice president. These joint activities and the various interfaces are described in procedures, the Station Administrative Orders (SAOs) and OP-290-1.

**Element 2** determines the scope of the modification. A meeting to review project scope is held. Engineering field walks are conducted to resolve conflicts concerning proposed plant configuration. A design review is held to approve the final conceptual design scope document. An additional field walk with the appropriate disciplines is conducted based on sketches and drawings.

**Element 3** provides that detailed engineering and design be conducted to prepare a technical package. Modification procedures are prepared with review by the cognizant engineering section manager(s). The Discipline Engineer prepares the design criteria for a modification and completes an extensive list of questions and considerations contained in an exhibit. These questions cover appropriate design and program interface. The Design Criteria contain responses to all applicable design considerations outlined in the exhibit. The Discipline Engineer also reviews the Design Bases Documents (DBDs) and other pertinent plant documents for existing criteria. In developing the criteria, the Discipline Engineer considers Classifications or Safety Evaluations previously issued (or pending) and obtains a list of modifications to the equipment or structure involved from the IP 2 "Modification and Calculation Indexing System" for review.

The Design Criteria are reviewed by cognizant section managers. The final Design Criteria for Class A (which includes safety-related items), FP (Fire Protection) and MET (Meteorological) jobs receive an independent design verification. The final Design Criteria are approved by the appropriate section managers and distributed in accordance with standard distribution.

The OP-290-1 procedure, Preparation and Review of Detailed Designs, establishes the requirements for the preparation and review of detailed design and as-built drawings.

Section 5.12 of OP-290-1 assigns responsibility for preparing the modification package to the Discipline Engineer, defines several types of Modification procedures (Major, Minor, Set Point Only, Generic) and for equivalent replacement modifications, a Determination Of Equivalency (DOE) and describes the process for preparation, review, and approval of each.

Discipline Engineer responsibilities include preparing and establishing requirements for the modification. Requirements include developing a scoping document, implementing procedures for a Safety Impact Questionnaire (SIQ) and Safety Evaluation (SE), initiating requests for drawings and specifications, and assessing the impact on other programs and documents.

The plant modification package includes the engineering documents necessary to fully describe a change to the plant configuration.

The Modification Procedure addresses configuration control processes. The Modification Procedure cover sheet contains check points and fields for data from many of the above items in the procedure, including: Classification, SE, Safety Assessment System Change, Simulator Change, Penetration of Fire Barrier, CCR (central control room), Human Engineering, System Description Change, Setpoint Change, EQ Equipment/Requirements, Loop Sketch Change, Essential Corrective Action, Tag Number Management System (TNMS) Change, UFSAR Change, ASME XI Quality Group Exempt/Outside Boundaries, and Weld Requirements.

The Preliminary Modification Package is transmitted to the engineering section managers as appropriate, and the Manager, System Engineering and Analysis, who is responsible for coordinating the plant review in accordance with SAO-405, Modifications to IP 2 Facilities. A Detailed Design Review/Comment Resolution Meeting and field walk are normally conducted (Reference OP-290-1, Section 5.12).

The Modification Package receives an independent design verification in accordance with OP-290-1 Section 5.13 before the Final Package is issued.

**Element 4** provides safety classification of equipment and the required SEs.

Section 5.14 of OP-290-1, Preparation, Review, and Issuance of SE, Classifications, and Part 21 Determinations, describes the engineering procedures required to implement the requirements of 10 CFR 50.59 in the design process. This procedure interfaces with the station procedure, SAO-460, Safety Evaluations. The preparation of SEs is described in Section 3.1.4.1 below. The procedure, contained in both SAO-460 and OP-290-1, covers preparing, reviewing, and issuing SEs and processing Part 21 Determination requests under 10 CFR 21. It also covers requests for Classification of equipment or work in accordance with the general listings of Non-Class, Class A, FP, and MET systems in CI-240-1. A SE or, in some cases, a SE Screening for jobs with lower safety significance is prepared for all IP 2 jobs assigned to Engineering regardless of classification. A Part 21 Determination must be completed for all Class A projects.

The Section 5.14 procedure includes a SIQ that is completed by the Discipline Engineer and reviewed by Nuclear Safety and Licensing (NS&L). The SIQ requires explanations or discussions on 26 general questions exploring appropriate safety-related areas that may be impacted by the modification. The procedure also provides definitions for Safety Impact Levels and directions for preparing SE documents.

If there are changes to the design criteria or scope of the project after the SE or Screening Checklist has been approved, the Discipline Engineer evaluates the safety impact of the change to determine whether a review by NS&L is required. If it is determined that the change is such that a revision to the existing SE or Screening Checklist is warranted, a Change of Scope form is issued for review. NS&L, determines whether the existing SE or Screening Checklist is still valid.

In **Element 5** the engineering review work results in a Design Review.

The detailed drawings for plant modifications require a thorough drafting and content review by a design checker using written standard procedures. This review is documented and the identification of the checker clearly indicated on the drawing.

All drawings prepared for Class A, FP, or MET systems or components require an independent design verification. This is performed and documented in accordance with Section 5.13 of OP-290-1. Design calculations supporting the detail design are checked or verified, depending on the classification, in accordance with Section 5.13 or Section 5.16 of OP-290-1.

After the reviews are completed, a drawing package, consisting of completed, but unapproved drawings, is given to the Discipline Engineer for review. After engineering comments are resolved and those accepted are incorporated into the drawings, the drawing package is issued to the user or Constructor for comment as part of the Preliminary Modification Package. Prints of parent drawings used for the Design Modification Drawings (DMDs) are included in comment packages. After comments are resolved or incorporated into the design drawings, the Discipline Engineer and other appropriate representatives participate in a field walkdown of the installation package. This is usually conducted in association with the required Detail Design Review (DDR) meeting. The results of the field walk are documented and included in the Project File.

**Element 6** requires reviews by SNSC and reviews of the Project File and Project Modification Procedures.

The Section Manager gives final engineering approval of the Modification Procedure after the SE document is approved. Final approval of the Modification Procedure is by the Manager, System Engineering and Analysis.

The Discipline Engineer presents the modification package to SNSC, if required.

A Project File Review Team (PFRT) performs a Completeness Review for major modifications, using the Completeness Review Form as a guide at the Project File Review. PFRT consists of a Section Manager or his designee, who is not involved with the design of the project under review and invited engineers and managers of other disciplines. The purpose of the review is to assure completeness and consistency of the project documents supporting the plant modification and to confirm compliance with engineering procedures and practices. This includes review of documents (e.g., design criteria assumptions, calculations and material specifications) and conformance to the 10 CFR 50.59 safety review or evaluation and to the applicable codes, standards, and additional regulatory requirements. The scope of this review covers the engineering modification process under OP-290-1 and other quality assurance practices and procedures for the preparation, control, and documentation of the modification package to the point of engineering release to the field. A checklist of PFRT review elements is completed and maintained in the project review file.

Distribution of the modification package is specified in Section 5.8 of OP-290-1.

**Element 7** is construction and the Report of Installation. Construction of the modification is monitored by the discipline engineer who also participates in cost and schedule resolutions, makes revisions if justified and approved, and reviews site construction. A Report of Installation (ROI) is issued by the Project Managing Authority (PMA) upon completion (or partial completion) of modification implementation in accordance with SAO-405. The system engineer prepares the startup authorization. PMA provides the Report of Installation.

**Element 8** closes out documentation and provides project feedback.

Upon receipt of an ROI for completion or partial completion of plant modifications, the plant documents are updated to reflect the as-built conditions and the applicable drawings are revised. The Discipline Engineer is responsible for assembling all original DMDs for the modification and coordinating revision of related design documents, including the UFSAR. Approximately 1 year after installation post implementation review is performed to critique major modifications and review lessons learned.

### 3.1.2 Design and Configuration Control Programs - Current

#### 3.1.2.1 Drawing Management Control System

The Drawing Management Control System controls the engineering design process for drawings and supports the primary design procedure for modifications, OP-290-1 and other lower tier documents provide guidelines on format and on preparing or revising DMDs and parent drawings. Identification of classification on drawings is detailed in these guidelines. They reasonably assure that the contents of the master drawing are proper and support design bases information in the as-built process. The DMDs are assigned a unique sequential number controlled by the DMCS (Drawing Management Control System). These procedures and DMCS help individuals to access the reference to other DMDs for each drawing and to review and determine any impact of the change on the design bases.

Design bases are also maintained under the requirements of station procedure SAO-522, Control of Drawings and Drawing Change Information. According to this procedure, all station personnel involved in the planning and execution of tests, inspections, operations, maintenance, repairs, or modifications to the station are required to use controlled copies of drawings. This reasonably assures that the plant design bases are maintained.

The drawings provide references to the equipment, structure and system design parameters, and materials configuration, e.g., the piping lines are uniquely identified with reference to the design specification. This information is used in determining the system requirements and incorporating requirements to the change, reasonably assuring that the design bases are maintained. Similarly, other references to design requirements on drawings are with reference to vendor drawings applied during changes to the system, structure, or component.

Quality Assurance requirements are also identified on drawings and are administered under OP-290-1 and lower tier documents. Classification of DMDs is determined by OP-290-1. The classification requirements are applied during installation of a change and are transferred to the master drawing, maintaining the design bases.

#### 3.1.2.2 Engineering Assurance Program

In 1989 the Engineering Assurance (EA) Program was established to affirm quality and excellence in engineering and design products through a continuous process of self-assessment to identify areas of weakness and to implement corrective actions. This program is conducted under the Chief Engineer, Nuclear Power Engineering. Its objective is to provide overview of the engineering and design processes and to facilitate implementation of the EA program to achieve quality improvement and effectiveness. The program also provides feedback to line organizations and a means to self-assess potential areas for improvement. The EA program reviews and evaluates the effectiveness of the engineering and design processes, assesses the quality of engineering design products and performance, identifies areas of potential weakness and determines root cause, and provides feedback to engineering management for corrective actions.

#### 3.1.2.3 Environmental Qualification Program

The Environmental Qualification (EQ) Program ensures that EQ criteria are appropriately applied to equipment regulated by 10 CFR 50.49 and describes the methods that demonstrate and maintain this qualification. The EQ Program incorporates NRC regulations, Regulatory Guides, and positions and guidelines, as well as IEEE standards and sound engineering practices. This

EQ Program is applied to all phases of design and operation for IP 2. The program includes the quality assurance requirements necessary to assure compliance with 10 CFR 50, Appendix B.

The EQ Program applies to all EQ activities of Con Edison personnel and support personnel associated with:

- development of and revision to documents in the EQ Central File;
- EQ activities associated with the design, procurement, installation, testing, surveillance, maintenance, and modification of EQ equipment;
- training personnel in EQ Program requirements;
- auditing the EQ Program; and
- reporting and resolution of non-conformance or deficiencies.

The EQ documents and associated files are design documents which provide direct support of the design bases. They define the environmental parameters and equipment qualification requirements in accordance with 10 CFR 50.49.

There have been numerous audits of the EQ Program over the years. A Con Edison Nuclear Quality Assurance, EQ program Audit was performed in 1995 (Report No. 95-08-E). This audit evaluated the implementation of the EQ program for compliance with 10 CFR 50.49. Emphasis was placed on evaluating the EQ files and the effectiveness of the EQ program controls in maintaining the continued qualification of installed equipment. Various types of equipment and EQ files were selected for audit with respect to operating requirements, supporting qualification documentation, installed configuration and the maintenance of qualification through the plant modification and maintenance processes. The selection basis emphasized equipment recently replaced in the plant because its qualified life had expired, equipment that traditionally requires periodic replacement due to environmental considerations, equipment that typically exhibits EQ-related configuration requirements because of uniquely tested configuration, and any new equipment installed that had not been previously audited. Follow-up reviews of EQ file completions by the PFRT (see Element 7 or Section 3.1.1.2), after plant modifications are implemented provide further assurance that design basis information related to Environmental Qualifications is properly maintained.

The EQ Program was found to be in compliance with the requirements of 10 CFR 50.49. The audit concluded that the EQ Program is in compliance with applicable Con Edison policies and procedures and that sufficient procedures and controls exist to effectively implement the program.

### 3.1.2.4 Nuclear Safety Classification of Systems, Structures, and Components

The nuclear safety classification process is defined in SAO-401, Nuclear Safety Classification of Components, Parts, and Commodities. Nuclear Safety Classification is a key process used for maintaining the plant System, Structures and Components (SSC) design and configuration consistent with the design bases.

Corporate Instruction CI-240-1, the procedure governing Con Edison's Quality Assurance Program, provides a list of Class A systems and components. CI-240-1 recognizes that not every portion of each of the listed systems and components contributes to preventing or mitigating the consequences of postulated accidents. Therefore, an individual component, part, or commodity may be evaluated to determine its appropriate nuclear safety classification. Evaluations under SAO-401 may change the classification.

SAO-401 describes the process for reclassifying items that have previously been classified or for classifying items that are installed in the plant but do not have a classification. The process requires the identification of reference documents used to provide the basis for classification. By assessing the impact of the failure of an item on the ability to maintain required safety functions and by linking this to documents such as the UFSAR, Technical Specifications (TS) or DBDs, the process reasonably assures that classifications are performed consistent with the design bases.

The nuclear safety classification process identifies items subject to the Quality Assurance Program. The completed classification documents provide a rationale for concluding that SSC classification is consistent with the design bases.

### 3.1.2.5 Seismic Qualification Program

In February 1987, the NRC issued GL 87-02, Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46. This GL encouraged utilities to participate in a generic program to resolve the seismic verification issues associated with USI A-46. As a result, the Seismic Qualification Utility Group (SQUG) was formed and it developed the "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment." In May 1992, the NRC issued GL 87-02, Supplement 1, the NRC review of the GIP, including Supplemental Safety Evaluation Report 2 on the GIP.

A report was written to describe the results of the seismic reviews performed to resolve USI A-46. USI A-46 work was performed by Con Edison personnel and outside engineering consultants. Safe shutdown paths have been identified to ensure that essential safe shutdown functions can be accomplished following a safe shutdown earthquake. The equipment required to accomplish safe shutdown functions was established based on the identified safe shutdown paths. A Composite Safe Shutdown List was prepared as specified in the GIP. In addition, separate Seismic Review and Relay Review Safe Shutdown Lists have been prepared as subsets of the Composite Safe Shutdown List. The report documents the results of the safe shutdown equipment identification, the screening verification and walkdown, relay evaluation, tank and heat exchanger review, cable and conduit raceway review, and outlier identification and resolution. The report of this program was submitted to NRC December 31, 1996.

#### 3.1.2.6 Fire Protection Program Plan

The Fire Protection Program Plan (FPPP) describes the IP 2 Fire Protection Program, incorporating the various commitments and descriptive information about the program that have been provided to the NRC. It is additionally intended to provide a single reference document for information pertaining to the program and is the focal point for controlling Fire Protection Program activities.

The FPPP has been prepared to satisfy the requirements of 10 CFR 50.48(a) for development and implementation of a Fire Protection Plan. In addition to providing programmatic controls, the FPPP may also be used as a reference document to aid in program reviews, audits, and in the review of proposed modifications, design bases, and proposed changes to procedures.

#### 3.1.2.7 American Society of Mechanical Engineers (ASME) Section XI

The responsibility for developing, revising, and maintaining the ISI Program currently rests with Con Edison's Nuclear Quality Assurance (NQA) Department. This responsibility includes the development, review, and maintenance of the associated relief requests. NQA initiates changes to the ISI Program Document and coordinates review of the changes with affected organizations.

IP 2 was designed prior to the development and implementation of ASME XI. Accordingly, ASME XI was not part of the original plant design requirements. When implementation of ASME XI was required, ISI Programs, including relief requests, were developed and submitted to the NRC. Today , the ISI programs have the practical effect of adopting ASME XI as a standard except where relief requests identify exceptions to the ASME Code.

ASME Code Relief Requests for the Inservice Inspection Program are described in Indian Point 2 Inservice Inspection Program, Third Interval, which was submitted to NRC for approval under of 10 CFR 50.55a. Such a submittal is required when the ISI Program is periodically updated to reflect later ASME Code Section XI requirements, generally every 10 years. The current ISI program, for the ten year interval, July 1, 1994 through June 30, 2004, applies the 1989 edition, without addenda, of the ASME Section XI Code to this interval.

### 3.1.3 Design and Configuration Control Programs - Historical

#### 3.1.3.1 Engineering Procedures and Design Drawing Upgrade

As part of the procedures upgrade, modification packages began to use a new DMD process in 1985. The objective was to eliminate the addition of jagged bounded areas to design drawings to identify pending plant modifications. A number of pending modifications resulted in the design drawings having many, sometimes overlapping, bounded areas associated with different modifications. This caused difficulty is reading the drawings.

All modifications currently use DMDs in lieu of revised design drawings. The previously issued modifications were not back-fitted; however, all then-pending modifications were reviewed to determine whether they were still required. Those that would be implemented were re-evaluated and upgraded to the 1985 procedure requirements. For those that were canceled, the design drawings were restored to the pre-modification configuration.

With the DMD process, separate drawings are developed and used for modifications. The parent drawings are not altered until a modification is implemented (See 3.1.1.2 above). The parent drawings represent the plant configuration.

In 1989 engineering procedures governing modification processes were upgraded providing a number of improvements to the engineering process. The improvements enhanced clarity and more fully addressed the completeness and consistency of documentation, interfaces and reviews for modification packages. Over the years, a number of improvements have been made to the modification process. Some of these include:

- Review and enhancement of Engineering Operating Procedures (OP-290-1), to eliminate fragmentation and provide more consistent information;
- Addition of flow charts and check lists to the Engineering Operating Procedures;
- Enhanced design verification process;
- Addition of specific Engineering Operating Procedures for EQ, ASME XI, Fire Protection/Appendix R, CCR Human Factors Engineering, and Simulator Configuration Control;
- Development of a computerized modification procedure index and a calculation index; and
- Use of a project priority heavily weighted by Probabilistic Safety Assessment (PSA).

3.1.3.2 Modification Packages Review

From 1988-90, a review of the IP 2 modification packages and files was conducted. Its purpose was to enhance modification packages and project files by making them more complete. The effort was started as part of the Configuration Management System initiative undertaken by Con Edison.

A contractor, Burns & Roe, entered modifications into the modification tracking system and reviewed approximately 1,300 modifications and files to determine their completeness. A retrieval effort was undertaken to collect the information identified as missing from the individual modification files. The significance of this undertaking is that project files were made more complete and information was centralized and made more easily retrievable. There were approximately 3,500 individual enhancements to the files as a result of this initiative.

3.1.4 Safety Evaluation and Safety Analysis

3.1.4.1 10 CFR 50.59 Safety Evaluations

IP 2 compliance with 10 CFR 50.59 is controlled by Station Procedure SAO-460, 10 CFR 50.59 Safety Evaluations. This procedure delineates the requirements for the preparation and documentation of safety evaluations for design modifications (including equivalent replacements, setpoint changes, temporary repairs, and jumpers) and for certain new procedures and changes to procedures. Guidance from such documents as the NRC Inspection Manual on 10 CFR 50.59, NRC IE Circular 80-18, and NSAC-125 is incorporated in SAO-460.

Engineering design procedures in the Engineering Operating Manual OP-290-1 incorporate the requirements of 10 CFR 50.59. These procedures cover preparing, reviewing, and issuing SEs and processing Part 21 Determination requests under 10 CFR 21. They also cover requests for classification of equipment or work in accordance with the general listings of Non-Class, Class A, FP, and MET systems. A SE or, in some cases, a SE Screening for design engineering of lower safety significance, is prepared for design modifications assigned to engineering regardless of classification. A Part 21 Determination is also completed for all Class A projects.

For changes requiring a SE, the SE is the permanent document that explains why a change does not involve a USQ. The procedure requires that the SE include specific and sufficient information to be a stand alone document addressing such questions as (1) What design documents could be affected? (2) Which design criteria are met and how? (3) Which regulatory requirements or commitments are involved? Consideration of the design and licensing bases for the facility is an integral part of the process.

A SIQ ensures that a predetermined set of questions is considered and answered to help determine the level of detail required to analyze the change and to facilitate the required documentation. The preparer is required to identify and consider failure modes introduced by the change and their possible effects on the functions of safety-related systems, structures, or components. The preparer of an SE is required by this procedure to make a Safety Impact Level determination. The process ensures that the SE documents the specific thought process, with an appropriate level of detail for that change, and supports the logical conclusion that no safety function, or margin of plant safety, would be degraded by this change. In so doing, it supports the maintenance of the plant within its design bases. The procedure acknowledges whether the UFSAR is impacted. This information is then later used to support updating the documents as required by 10 CFR 50.71(e) and a summary of approved plant changes as required by 10 CFR 50.59(b)(2).

The effectiveness of SAO-460 is assessed by internal audits conducted by the NQA Department. These audits have identified findings and observations for improvements that are addressed in formal updates to the procedure, followed by training of individuals as appropriate. All changes to SAO-460 are made consistent with the requirements of the SAO-100, Indian Point Station Procedure Policy, procedure revision process, which requires that appropriate station reviews or approvals are conducted prior to issuance.

3.1.4.2 UFSAR Program

10 CFR 50.71(e) requires a licensee to periodically update the UFSAR. These updates are to include the effects of changes made in the facility or procedures described in the UFSAR as well as SEs performed in support of requested license amendments or conclusions that changes did not involve an USQ (10 CFR 50.59 process).

IP 2 complies with this requirement by requiring an acknowledgment of impacts to the UFSAR for all SEs performed in accordance with 10 CFR 50.59. This process is currently controlled by SAO-460, 10 CFR 50.59 Safety Evaluations and is applied to the above considerations for UFSAR updates. This requirement of SAO-460 is replicated in OP-290-1, which was recently updated to require inclusion of the impacted UFSAR pages in the modification package.

When it is determined that the UFSAR is impacted in the process of performing a SE, the procedure requires that this impact be identified in the SE. Upon approval of the SE, a copy of the approved document (and annotated page(s)) is provided to an individual in Nuclear Safety and Licensing (UFSAR coordinator) for incorporation in the next update of the UFSAR. The responsibility for the effort is assigned to a senior level person with considerable nuclear plant experience, and the process includes other organizational reviews and approvals of the updated document, prior to issuance. Additionally, discovered discrepancies or inconsistencies in the updated UFSAR are normally communicated directly to this individual or documented in the station's corrective action process which assigns and tracks resolution. Other than minor editorial changes, all changes to the UFSAR are documented with a SE in accordance with 10 CFR 50.59.

Con Edison has initiated a program to review the IP 2 UFSAR. The program includes a review of the effectiveness of the processes and procedures used to keep the information in the UFSAR accurate and up-to-date. The objective of the UFSAR Review Program is to (1) provide reasonable assurance that the UFSAR information is correct and consistent with the plant's design bases information by identifying and resolving inconsistencies and making the necessary changes in the UFSAR and/or plant controlled documents/plant physical configuration; and, (2) to insure that the controlling procedures that keep the UFSAR accurate and up to date are effective. The controlling documents that describe the details of the UFSAR Review Program are the Indian Point 2 UFSAR Review Program Plan, dated November 1996, and System Engineering Procedure SE-S-12.400 Identification, Tracking, Control and Resolution of Discrepancies Resulting from the UFSAR Review, dated October 1996. Combined corporate-site teams have been established and are performing this review. This effort may be expanded to include contractor support.

As a result of the preliminary FSAR review we initiated in the later part of 1996 and our efforts in preparing the response to this information request, we are expanding the scope of our FSAR review program as described in Attachment A, Commitments.

3.1.4.3 UFSAR Fuel Design Safety Analysis

Since 1973, Westinghouse has provided the analysis for the core designs and fuel fabrication for Con Edison. The safety analysis inputs and results are maintained by Westinghouse. Westinghouse reviews changes for any impact on the safety analyses as requested by Con Edison. Westinghouse also updates the safety analyses for any core or fuel design changes. All safety analyses are controlled by Westinghouse procedures. Con Edison reviews plant changes for impact on the safety analyses and takes appropriate actions such as performing engineering, safety evaluations, and contacting Westinghouse as mentioned above. Con Edison also maintains and makes necessary changes to the UFSAR and plant procedures.

The design of the core involves a detailed series of steps performed by Westinghouse and Con Edison. Cycle design at Con Edison considers the requirements and assumptions for the next cycle design. When the cycle design requirements and assumptions are finalized, they are included in a letter to Westinghouse requesting a set of fuel management cases for the next cycle. This fuel management plan includes a base case and additional cases used for evaluations of options. Westinghouse runs these cases and provides a fuel management report for Con Edison's evaluation. Con Edison evaluates the results for accuracy, performs checks of the data, and chooses one case for the cycle design. This is transmitted to Westinghouse in a letter containing the final energy and core design requirements for the next cycle, along with schedule requirements and deliverables, fuel products features, anticipated TS changes, and any other constraints or requirements specified by Con Edison. Westinghouse later provides the preliminary core design which Con Edison independently checks. When the preliminary design is accepted, the uranium and enrichment requirements are determined and the number of assemblies needed to be fabricated is ordered.

The Reload Safety and Licensing Checklist is completed by Con Edison and sent to Westinghouse. This provides the energy requirements, plant modifications completed and planned, safety evaluations performed, and assumptions used for the reload design. The checklist is required by Westinghouse to initiate the final core design. After Westinghouse receives the checklist, a design initialization meeting is held within Westinghouse to discuss the information in the checklist and resolve any outstanding issues. Con Edison attends this meeting. After this meeting, each group within Westinghouse begins to perform its reload design function. Westinghouse generates the Reload Safety Analysis Checklist (RSAC), which compares the reload and current safety parameter limits, to ensure that the reload values are within the current safety parameters. If the reload values are not within the current safety parameters, an evaluation is performed, the safety analysis is revised, or the core design is changed.

Westinghouse's core design responsibility is to determine a core loading pattern that is acceptable to Westinghouse and Con Edison. Con Edison models Westinghouse's final loading pattern to ensure that assemblies are properly placed and have correct burnups, and an assembly power distribution comparison is made to confirm power peaking. Con Edison performs additional checks for boron and control rod worths, axial power distributions, and boron letdown.

Westinghouse performs the Reload Safety Evaluation (RSE) to ensure that the core design meets the criteria for core reload safety analysis. The RSE is sent to Con Edison for review and finalization prior to core loading and operation. Westinghouse also provides the k-effective and shutdown boron requirements during refueling.

Prior to core loading and core operations, Con Edison performs SEs of the core loading and core operations to ensure that safety questions have been addressed and resolved. Core operations include startup and subsequent core operations. Using the RSE, Con Edison develops a SE and obtains approval by the SNSC.

Before startup, Con Edison receives a Westinghouse package that contains data needed to verify the core design during initial startup after refueling to ensure that the plant is operating as designed and therefore that all safety parameters are met. Westinghouse provides Con Edison with the Nuclear Parameters and Operations Package (NUPOP), which summarizes the parameters required for monitoring and operations of the core. Con Edison performs comparisons using in-house codes to verify pertinent parameters from the NUPOP, such as power peaking factors, boron worth, and control rod worths.

The most important aspect of core design is the final verification that the core is designed as planned and detailed in the NUPOP and that the core is loaded as planned and detailed in the core refueling procedures. Startup testing is a necessary, but not sufficient condition to determine whether the core is built and loaded as designed. Therefore, Con Edison continues to monitor the core during core life to ensure that core performance is as expected by the core design.

Core design work performed by Westinghouse and Con Edison is controlled by procedures. The core design basis information is maintained by Westinghouse. The fuel design is developed and licensed by Westinghouse, and the design bases information is maintained by Westinghouse. Con Edison reviews the design information and makes any necessary changes to address the fuel design.

3.2 Request

**(b) Rationale For Concluding That Design Bases Requirements Are Translated Into Operating, Maintenance And Testing Procedures**

Response

Con Edison is confident that the IP 2 design bases requirements are translated into operating, maintenance, and testing procedures. These procedures are governed and controlled by Station Administrative Orders (SAOs) and departmental and section administrative procedures.

Station Administrative Orders provide station policy. The policy on procedures is specified in SAO-100, Indian Point Station Procedure Policy. SAO-100 requires comprehensive use of, and compliance with procedures. This requirement adds assurance that operations, maintenance, and testing activities that could affect plant design bases are accomplished in a controlled manner. Further assurance that design bases requirements are translated into operations, maintenance, and testing procedures is provided in SAO-460, 10 CFR 50.59 Safety Evaluations. Both SAO-100 and SAO-460 state that safety evaluations, as defined in 10 CFR 50.59, must be performed for procedures and procedure changes that may impact the Updated Final Safety Analysis Report (UFSAR). They also specify that any proposed procedure or procedure change that renders, or may render the UFSAR or subsequent safety analysis reports inaccurate, and those which involve or may involve potential unreviewed safety questions, be approved by the Station Nuclear Safety Committee (SNSC) prior to implementation.

SAO-206, Jumper Log, governs the jumper (temporary modification) process. To ensure that jumpers are used in accordance with the plant's design bases, SAO-206, provides that jumpers not be installed without a 10 CFR 50.59 Safety Evaluation. Control and elimination of operator workarounds is specified in SAO-135, Nuclear Power Policy Statements. Under this policy, every member of Nuclear Power (NP) is responsible for reporting workarounds to the appropriate level of management when they are discovered. Reporting a workaround and actions taken to address them provide reasonable assurance that workarounds do not adversely affect the design bases.

Corrective or preventive maintenance procedures, which keep plant equipment in a condition of good repair at or near original design and capable of performing their intended function, are specified in SAO-251, Conduct of Maintenance. Physical changes to plant systems, structures or components or replacements with other than like-in-kind are defined as modifications in SAO-251, and controlled by SAO-405. Modifications to Indian Point Facilities require application of SAO-460. SAO-405 provides instructions and administrative requirements for implementing modifications. Section 2.8 of the SAO-405 provides guidance for conducting walkdowns, pre-implementation meetings, and reviews of all modification packages for potential impact on existing procedures. When plant systems are modified, SAO-405 specifies the procedure for updating the design bases.

Plant activities are routinely reviewed and audited to determine that operation, maintenance and testing of the nuclear plant is in accordance with plant procedures. The results of these reviews and audits have identified no programmatic deficiencies in the integration of the IP 2 design bases into the procedures for operations, maintenance, and testing.

Plant Emergency Operating Procedures (EOPs) and other operating procedures each underwent a separate extensive upgrade in the early 1980s. A writer's guide was developed to provide consistency in procedure development and future revisions. The EOPs, other operating procedures and the policy and procedures that are in place to revise them provide reasonable assurance that design requirements are appropriately incorporated into the operating and emergency procedures.

Con Edison's procedure programs and reviews of those programs provide reasonable assurance of control and proper application of design bases information in the procedures used by operations, maintenance, and testing.

### 3.2.1 Operating Procedures Updates

#### 3.2.1.1 Emergency Operating Procedure Update

In late 1981, the Westinghouse Owners Group (WOG) determined that a major enhancement to emergency procedures could be accomplished using the Emergency Response Guidelines (ERGs). The ERGs are the written generic guidelines that contain explicit directions for control room operators to implement emergency response strategies. Information presentation and evaluation mechanisms have been used in the ERGs presentation to make them more useful under high stress conditions. ERGs provide detailed guidance on how to prepare and implement the EOPs, including EOP development, writing, and maintenance.

The ERGs were created using a simulator rather than accident analyses. Conceptually, the ERGs were to employ two complementary and interrelated guideline subsets (one event related, the other function related). The objective was two-fold: to provide prioritized operator guidance for recovering the plant from an emergency transient and to assure that the plant safety status was explicitly monitored and maintained during plant recovery. To address NRC concerns on smooth transitions between guidelines, the ERGs were developed in a human factored format that explicitly identified guideline transitions. A systematic evaluation of event sequences using probabilistic safety assessment (PSA) was performed to determine which guidelines should be included in the ERG set. These guidelines contained the following five elements:

- Optimal Recovery Guidelines
- Critical Safety Function Status Trees
- Critical Safety Function Restoration Guidelines
- Example Guideline Format
- PSA-based Procedures Evaluation

IP 2 uses a procedurally controlled method to convert the generic WOG ERGs into plant-specific EOPs. A multi-disciplinary team was and continues to be employed to incorporate changes to the ERGs into IP 2 EOPs. The Low Pressure (LP) version of the ERGs is the basis document for IP 2. Any differences between the ERG LP reference and IP 2 are identified in the IP 2 Design Differences Document. IP 2 plant-specific background documents were also created when the IP 2 EOPs were created. Any differences between the WOG Generic ERGs and the IP 2 plant specific EOPs are documented in the IP 2 Step Differences Document.

IP 2 procedures that govern maintenance of the EOP require: validation of EOP changes, normally performed on the IP 2 simulator; verification to ensure conformance to the EOP Writer's Guide; a safety evaluation under 10 CFR 50.59 for each EOP change; Station Nuclear Safety Committee (SNSC) review and approval prior to EOP implementation; and training for all licensed operators and appropriate non-licensed operators (if the revision affects actions outside the Central Control Room) as part of an EOP change.

The IP 2 EOPs are in regular use on the IP 2 simulator. Feedback is provided for procedure enhancements and, most importantly, it provides assurance that the EOPs maintain the capability to mitigate and control accidents and transients described in the UFSAR.

3.2.1.2 Operating Procedures Update

In the fall of 1982, Con Edison engaged a contractor, Stone & Webster, to perform an extensive review of all plant operating procedures, check-off lists (COLs), graphs, log books, Station Administrative Orders (SAOs), and Operations Administrative Directives. The program was undertaken to identify and correct inaccuracies, inconsistencies, and other deficiencies in the operating procedures. The program was completed in the spring of 1985. The UFSAR, Technical Specifications, NRC correspondence, as well as Westinghouse documents, setpoints, precautions and limitations, were reviewed. The need to validate setpoints, which were not traced to a design bases document, was also established. Source documents representing the IP 2 design and licensing bases were gathered and transmitted to Stone & Webster at the beginning of the project and during the course of their review. The review process identified operational commitments from licensing documents and tabulated them for incorporation in revised procedures. During the process, any inconsistencies or discrepancies among documents were referred to Con Edison for resolution as system review items, and their final disposition was incorporated into the updated procedures. Deficiencies that could have resulted in misoperation or delayed operator action were corrected. Discrepancies in procedures, drawings, and documents were identified and resolved. A writers guide procedure was also developed to ensure consistency in procedure development and future revision.

Based on the extensive level of effort devoted to upgrading the IP 2 EOPs and operating procedures, Con Edison has reasonable assurance that the operating and emergency procedures and policies adequately incorporate design requirements.

### 3.2.2 Operations

#### 3.2.2.1 Operating Procedures

The Generation Support Section at IP 2 is responsible for the development, maintenance, and revision of all Operations Section Procedures. These include Operations Administrative Directives (OADs), System Operating Procedures (SOPs), Plant Operating Procedures, (POPs), EOPs, Alarm Response Procedures, Abnormal Operating Instructions, Temporary Operating Instructions, and Check-Off Lists (COLs). These documents and procedures are administratively controlled by SAO-100, Indian Point Station Procedure Policy, and Generation Support Administrative Directive-9 (GSAD-9), Operating Procedure Development and Control.

In accordance with IP 2 TS Section 6.8, Station Administrative Orders (SAOs) have been developed to provide station administrative policy. Both SAO-100 and SAO-460 provide the administrative guidance necessary to ensure that design bases requirements are translated into operating procedures. Both of these administrative orders state that Safety Evaluations, as defined by 10 CFR 50.59, must be performed for procedures and procedure changes that could render the Updated Final Safety Analysis Report (UFSAR) inaccurate. The UFSAR must be reviewed when developing or changing a procedure to determine whether the procedure or procedure change might result in an unreviewed safety question.

The Generation Support Section revises existing or generates new operating procedures as applicable. SAO-405, Modifications to Indian Point Facilities, provides instructions and administrative requirements for the implementation of modifications. SAO-405 Section 2.8, stipulates that the Generation Support Section is requested to participate in walkdowns and pre-implementation meetings and review all modification packages for potential impact on existing procedures. This further assures that physical changes to the facility are translated into plant operating procedures.

#### 3.2.2.2 Design Changes and Jumpers

The following SAOs control the process which evaluates and reasonably assures that jumpers do not adversely impact compliance with the UFSAR.

SAO-206, Jumper Log, governs the jumper process and defines a jumper as the defeat or other alteration of a particular circuit, interlock control, or piping arrangement defined by plant design. Jumpers exclude the operation of installed defeat, bypass test, or other similar switches or devices that are part of the plant design. To ensure that jumpers are used in accordance with the plant's design bases, SAO-206, Section 4 provides that jumpers not be installed without a 10 CFR 50.59 Safety Evaluation.

A jumper must undergo a documented review according to SAO-460 prior to installation unless it is required for the immediate protection of the plant or its personnel. According to SAO-460, the preparer of the Safety Evaluation must determine whether the jumper represents a change to the UFSAR. The UFSAR must be reviewed to answer this question. The review and safety evaluation process give reasonable assurance that the design bases are not compromised by the use of jumpers.

3.2.2.3 Operator Workarounds

SAO-135, Nuclear Power Policy Statement Number 15, governs the control and elimination of operator workarounds. By definition: "A workaround is a deficiency in plant equipment, procedures or training that prevents a component from performing its intended function which has an impact on the margin of plant safety, personal safety or plant reliability." Based on recent operational events in the nuclear industry, it was recognized that a more formal approach was needed to control and eliminate operator workarounds. Under the IP 2 policy statement, every member of NP is responsible for reporting workarounds to the appropriate level of management when the workarounds are discovered. The appropriate manager reviews the workaround and brings it to the attention of the Daily Management Review Group (DMRG). The DMRG as described in SAO-132, Analysis of Station Events and Conditions, then categorizes the deficiency based on significance. In addition, the Manager of Operations Training reviews all workarounds to determine whether simulator or classroom training is required to address the problem.

Operator Workarounds are placed in the appropriate corrective action system (i.e., open item report, work request) as needed to followup on the deficiency. Based on this policy statement and followup actions, there are adequate reviews and actions to reasonably assure that workarounds are corrected.

3.2.3 Maintenance

3.2.3.1 Maintenance Work Control

Maintenance at IP 2 is governed by a SAO-251, Conduct of Maintenance, and is defined in this document as corrective or preventative work activity which keeps plant equipment in a condition of good repair at or near original design and capable of performing its intended function. This includes like-in-kind replacement. This definition limits routine maintenance to activities which are not changes to the existing design. When equipment becomes obsolete, and can not be replaced with like-in-kind, or when equipment changes are designed to effect enhancement, the modification process must be used. Physical changes to plant systems, structures or components, or replacements with other than like-in-kind, are defined as modifications in SAO-251 and are subject to additional controls. Such controls govern permanent modifications and setpoint changes, as well as temporary repairs, and temporary equipment installations or jumpers. A common element in each of these processes, is that the proposed change must be screened for safety impact in accordance with another Station Administrative Order, SAO-460, 10 CFR 50.59 Safety Evaluations. This document requires a Safety Evaluation to be performed for any alteration from approved plant design or method of operation by means of a modification, setpoint change, procedure or temporary repair to a plant system, structure, or component if it either:

a. causes a change in the UFSAR text or drawings,
b. causes a change below the detail of the UFSAR that would affect the UFSAR description of any equipment design, performance, function, or method of performing the function, or
c. affects other documents considered to be part of the licensing basis, including responses to generic letters, bulletins, and other licensing correspondence.

3.2.3.2 Maintenance Procedures

The plant TS provide the fundamental requirements, to establish and control written procedures and administrative policies in Section 6.8. Among other things, this TS requires an Administrative Control Procedure (SAO-100) for procedures that ensure:

a. each proposed procedure or procedure change involving safety-related components and/or operation of same receives a pre-implementation review by the SNSC except in case of an emergency,

b. each proposed procedure or procedure change which renders or may render the UFSAR or subsequent safety analysis reports inaccurate and those which involve or may involve potential unreviewed safety questions are approved by the SNSC prior to implementation, and

c. the approval of the Nuclear Facilities Safety Committee shall be sought if, following its review, the Station Nuclear Safety Committee finds that the proposed procedure or procedure change either involves an unreviewed safety question or, if it is in doubt, whether an unreviewed safety question is involved.

The SAO-251 governing maintenance requires that all maintenance activities be conducted in accordance with approved procedures. SAO-100, Indian Point Station Procedure Policy, provides direction for how the required procedures are to be developed, reviewed, and controlled. This procedure ensures that the maintenance procedures receive the reviews required by the TS described above. If it is determined that proposed procedure revisions may involve potential unreviewed safety questions or may render the UFSAR or subsequent safety analysis reports inaccurate, a Safety Evaluation in accordance with SAO-460, described earlier, is required. Compliance with SAO-100 also assures satisfying requirements of Corporate Instruction, CI-240-1, Quality Assurance Program for Operating Nuclear Plants, and the Quality Assurance Program Description.

These administrative controls and NQA Audits and third party evaluations of maintenance activities provide reasonable assurance that plant maintenance activities are consistent with design bases.

### 3.2.4 Testing Procedures

The Test and Performance Section is responsible for writing, reviewing, and approving tests to implement the surveillance requirements of the IP 2 TS. The Test and Performance Section also prepares tests to return plant equipment to service following maintenance or modification. These tests are administratively controlled by the following:

- SAO-100, Indian Point Station Procedure Policy

- SAO-460, 10 CFR 50.59 Safety Evaluations

- AD-SQ-2.000, Site Services, System Engineering and Analysis, Independent Safety Review, and Radiation Protection Section's Writers Guide

- AD-SQ-2.002, Review, Revision, Approval and Distribution of Site Services, Systems Engineering and Analysis, Independent Safety Review, and Radiation Protection Section's Procedures

- TP-SQ-11.015, Surveillance Test Procedure Issuance and Review Process

- TP-SQ-11.016, Post Maintenance Test Program

- TP-SQ-11.017, ASME Section XI Inservice Test Program

- TP-SQ-11.018, ASME Section XI Inservice Pressure Test Program

The above procedures reasonably assure that the design bases documents of the plant are appropriately reflected in surveillance and testing activities. SAO-460 is also used to ensure that the revisions to test procedures do not invalidate the design bases conditions.

When plant systems are modified, the modification package identifies the required testing necessary to demonstrate that the SSCs are operable and within the design bases. The design bases or supporting design documents of the plant in that case are updated by the modification process in accordance with SAO-405.

When periodic surveillance tests are conducted, a satisfactory or unsatisfactory determination is made. The determination is based on the test procedure results and acceptance criteria. If the test is unsatisfactory, a Significant Occurrence Report (SOR) is written, the SOR is reviewed by the DMRG. Where appropriate an evaluation (SAO-132 report) of the test results, determination of root cause, and identification of corrective actions may be required. SORs are tracked using the Condition Identification and Tracking System (CITRS) data base. During the performance of a test, if conditions are found that do not otherwise cause a test failure but require some correction, a work order or OIR is written to address the condition.

Through the use of the above procedures, NQA audits, and independent assessments, Con Edison believes that plant maintenance and testing procedures reflect the design bases provided by the engineering design and configuration control process described in Section 3.1.1.

3.2.5 SSFA and NQA Audits

In 1987, Con Edison instituted a program of internal Safety System Functional Assessments (SSFAs). SSFAs are in-depth, vertical slice, evaluations of the design, operation, maintenance, testing, and related support activities (e.g., training, material condition) of selected safety-related systems at IP 2. These assessments have historically involved a team of outside consultants with particular expertise in the area being audited. Twenty-one such assessments have been performed (See Section 3.3.2.1) to date, and the SSFA process is intended to be continued. These assessments have served an important role in assessing the integrity of safety-related systems. These SSFAs have identified some areas for follow-up corrective actions and some strengths.

The SSFA vertical slice approach not only considers the integration of the design bases into operations, maintenance, and testing evolutions, but also determines the configuration of SSCs and their performance in relation to the design bases. The SSFAs completed to date demonstrate the major efforts expended by Con Edison in recent years.

In addition to these SSFAs, there are additional internal audits conducted by the NQA Department. Many of these are mandated by the IP 2 TS to confirm that activities performed in accordance with the TS are performed correctly. Other internal audits concentrate on areas of importance such as Fire Protection (FP), Environmental Qualification (EQ), and Seismic Qualification (SQ). Unlike the SSFAs, which may be performed with a team of five to seven people, the internal audits are performed by one or two individuals from the NQA staff, sometimes supplemented by outside consultants. These audits constitute a factual assessment of the plant activities audited. They also examine aspects of operations, maintenance, and testing. Audits in these areas for the past two years have been reviewed, and it has been concluded that the issues raised, when resolved, do not repeat themselves. In general, the audits related to maintenance and operations have not resulted in findings or observations that relate to design bases issues. Extensive auditing of the test program is part of the internal audit program mandated by the Indian Point Unit No. 2 Technical Specifications (TS). These latter audits touch upon many aspects of the test program including scheduled performance, review of the test procedures to assure that they reflect the test objectives, corrective action in the event of test failures and anomalies, and verification that surveillance test results and surveillance intervals are valid.

The aforementioned procedures, reviews and assessments confirm Con Edison's belief that there is reasonable assurance for concluding that the design bases have been appropriately translated into operations, maintenance, and testing procedures. Any exceptions or significant issues, where further action is required, have been entered into the IP 2 corrective action system and are being tracked to resolution.

## 3.3 Request

### (c) Rationale For Concluding That System, Structure, And Component Configuration And Performance Are Consistent with the Design Bases

Response

Con Edison is confident that its processes and procedures are adequate to provide reasonable assurance that the IP 2 system, structure, and component (SSC) configuration and performance are consistent with the design bases, that personnel comply with these procedures, and that there is considerable evidence that the processes are effective. These processes are based on a multi-level approach to the management, control, and verification of the configuration and performance of the plant SSCs. The processes and programs which lead to reasonable assurance include:

1) Control of engineering, operations, maintenance, and test processes, including both the use of procedures and training in the use of procedures, which provide assurance that the design bases are properly considered;
2) Assessments, both internal and external, evaluate that the processes are properly used contributing to the assurance that the plant SSCs are consistent with the design bases;
3) Processes that support operations such as operability determinations, walkdowns, and testing programs, which provide additional assurance that the plant SSCs are consistent with the design bases; and
4) Additional specific initiatives and programs that provide verification of and contribute to the accuracy of the design bases and the consistency of the SSCs with the design bases by reviewing and upgrading existing design information or generating new information, as required.

The results of the implementation of these processes and programs provide confidence and reasonable assurance that IP 2 is operated and maintained within the design bases and deviations are managed by the plant corrective action systems. The detailed discussions in the following sections provide amplification of and support for this rationale.

### 3.3.1 Control of Processes

Engineering, operations, maintenance, and test processes are controlled by procedure. The fact that these procedures require that the design bases be properly reflected in the design process and operations, maintenance, and test procedures and that procedural adherence is required provides a foundation for management confidence that the desired results are being achieved. Section 3.1

contains a detailed discussion of the engineering processes at Con Edison and demonstrates how they are controlled to maintain consistency between the design bases and the plant systems, structures, and components (SSCs). Section 3.2, contains a detailed discussion of the operations, maintenance, and test procedures at Con Edison and examines how they are controlled to maintain consistency between the design bases and the plant SSCs.

#### 3.3.1.1 Design Control

Design bases information has been and continues to be incorporated into procedures and processes in accordance with OP-290-1 Section 5.0, Engineering Procedures for Operating Nuclear Plants. Section 3.1 provides details on how this process is accomplished and controlled.

Changes to procedures and processes for plant operations, maintenance, and testing are reviewed against the UFSAR, and plant modifications are reviewed under SAO-405 to assure that any required procedure changes are properly implemented. Section 3.2 provides details on how this process is accomplished and controlled. This contributes to reasonable assurance that the configuration and performance of the plant SSCs are consistent with the design bases.

#### 3.3.1.2 Use of Procedures

SAO-100, Indian Point Station Procedure Policy, requires the use of and compliance with procedures. This subject is discussed in detail in Section 3.1 with regard to the processes for control of engineering, design and station configuration and in Section 3.2 relating to operations, maintenance, and testing. The comprehensive use of procedures for work affecting compliance with design bases requirements provides assurance that modifications to the design and to SSCs are accomplished in a controlled manner and that the configuration of the plant is controlled.

#### 3.3.1.3 Training

Processes that deal with station design control, configuration control, operations, maintenance, support, and training are governed by plant procedures. In addition to the confidence gained by controlling station processes through procedures, confidence is provided by the fact that personnel are trained and, where required, qualified in the use of appropriate procedures as well as in their job functions. Training is provided to personnel in the use of the procedures required for performing their responsibilities and in key processes, such as performing 10 CFR 50.59 Safety Evaluations. Procedures require that personnel be trained prior to performing a function such as 10 CFR 50.59 Safety Evaluations.

Design bases training for operators at IP 2 occurs via several mechanisms. Plant systems training for both licensed and non-licensed operators includes the function and design characteristics of systems. System lesson plans contain learning objectives for the function and design characteristics of plant systems. These objectives, combined with examination questions, provide assurance that students possess the knowledge of design bases of the plant systems at IP 2. In addition, the procedures that govern the operation of plant systems are also taught during initial and continuing operator training.

Licensed operator candidates receive additional training on the design bases of plant systems during initial licensed operator training. Technical Specifications, (TS) and their bases, are taught during the plant systems phase. Therefore, when a system is covered, both its design characteristics and the applicable TS are presented. Transient and Accident Analysis is one of the course modules presented during initial licensed operator training. This course covers analyzed plant events and the expected system responses. The information within the UFSAR provides a significant reference for the information presented in this module.

Watch Engineers (Licensed Senior Reactor Operators) have an important role with respect to design bases maintenance. They are safety reviewers for 10 CFR 50.59 Safety Evaluations (SE) for Jumpers under SAO-206 (Jumper Log) and SAO-460 (10 CFR 50.59 Safety Evaluations). The safety reviewer is responsible for verifying that all pertinent failure modes and effects of the proposed change have been considered and, determining whether an Unreviewed Safety Question (USQ) is involved. Watch Engineers receive specific training on SE, a prerequisite to qualification as a safety reviewer.

Modifications to plant systems are curricula requirements for continuing training for both licensed and non-licensed operators. Alterations to plant systems resulting in changes to the way systems are to be operated are covered through applicable continuing training programs in accordance with the systematic approach to training. These modifications, along with any applicable procedural revisions, are reviewed with appropriate station personnel.

Engineers who perform modifications to plant design are enrolled in the Engineering Support Personnel (ESP) Training Program. This program provides knowledge of nuclear technology to achieve safe and reliable plant operation. Design bases information is incorporated in various training courses included in the ESP Program. A sample list of courses with content related to design basis and course length follows:

Plant Systems Design Basis Course (160 hours)
Nuclear Safety Awareness (50.59) (40 hours)
Design Basis Accidents (24 hours)
Nuclear Codes and Standards (16 hours)
Seismic Qualification (16 hours)
Fire Protection (12 hours)
Environmental Qualification (16 hours)

Indian Point 2 Modification Process (SAO-405) (4 hours)
Indian Point Orientation Course (12 hours)
Engineering Support Personnel Continuing Training (8 hours semi-annually)
Operability Assessment (4 hours)

This training inculcates personnel with the importance of maintaining the plant SSCs consistent with the design bases. Personnel are instructed in the correct sources of and processes for considering information in their jobs that could affect plant SSCs. The training described adds to Con Edison's confidence in the consistency between the plant SSCs and the design bases.

3.3.2 Assessments

Several types of assessments are conducted at IP 2, many of them initiated and conducted by Con Edison and some of them initiated or performed by external organizations. These assessments evaluate that the processes are conducted in a controlled manner and therefore, contribute to an assurance that the SSCs are consistent with the design bases. If these assessments identify discrepancies, they are tracked by corrective action programs, described in detail in Section 3.4 until the deficiencies are corrected or resolved, with the end result again contributing to assurance that any discrepancies between the SSCs and the design bases have been corrected.

3.3.2.1. Safety System Functional Assessment (SSFA) - Self Initiated Assessment

SSFAs are in-depth vertical slice evaluations of the operation, maintenance, modification, testing, design and related activities or features (e.g., training and system material condition) conducted to evaluate the operability of IP 2 safety-related systems and Unit 1 systems that support Unit 2. These Con Edison SSFAs are planned, conducted, and reported in a manner like that employed in NRC Safety System Functional Inspections (SSFI's).

The objective of the Con Edison SSFAs is to evaluate system operability, configuration and performance and to identify any aspects of activities supporting or otherwise impacting safety-related systems needing improvement and if so, to recommend appropriate corrective actions. Follow-up and resolution of identified improvement areas stemming from SSFAs are conducted by the NQA Department. Included in the objective of the SSFAs is an evaluation of system configuration parameters and functions under the license, TS, UFSAR, design bases, and associated documentation.

The Con Edison SSFA program was introduced to build on the NRC's initiation of the SSFIs vertical slice program in the mid 1980s. The Con Edison SSFA program began in 1987 with the SSFA (87-11-A) of the IP 2 Auxiliary Feedwater System. These assessments were planned, conducted, and reported by teams of Con Edison and contractor personnel. External consultants were selected based on their knowledge and expertise in the systems and functions being evaluated. An integral part of each assessment was the use of NRC Inspection Procedure 93801 as a guideline. This was the procedure followed by the NRC in its SSFIs, with one exception: the Service Water System Operational Performance Inspection (SWSOPI), where the NRC Inspection Procedure 2515/118 was followed.

The following are typical of areas evaluated during an SSFA:

1. Review of available design and licensing bases information, appropriate to the sample, to establish acceptance criteria;

2. Review of selected design documentation and plant modifications to assess conformance of the plant's configuration to the design and licensing bases;

3. Review of selected test and maintenance records to assess the adequacy of testing in determining performance capability under specified conditions and the adequacy of the maintenance program in assuring the ability of the systems and components to perform their safety functions;

4. Field inspection of critical physical attributes as necessary to support the assessment plan and as allowed by plant accessibility;

5. Review of Safety Classifications and Safety Evaluations of select portions of the system versus the requirements of Con Edison procedures and 10 CFR 50.59 requirements;

6. Review of selected abnormal, emergency, and normal operating procedures to evaluate technical adequacy, system performance capability, and manual actions required for potential scenarios; and

7. Review of selected operational records (LERs, SORs and Maintenance Work Orders) to evaluate the adequacy of root cause determinations, reporting requirements, corrective action programs, and the timeliness and prioritization of maintenance.

In these SSFAs, the teams used the definition of design bases as defined in 10 CFR 50.2. The proper translation of the design bases into specifications, drawings, procedures, and instructions is deemed to be a requirement of 10 CFR 50, Appendix B, Criterion III, Design Control. The plant licensing bases, as identified in the TS, the UFSAR, and commitments to the NRC, were included as part of the basis of these assessments.

The earlier SSFAs (e.g., Auxiliary Feedwater System, Residual Heat Removal System, and Service Water System) examined design bases documentation availability as a generic question. Table 3.3.2.1 lists the SSFAs performed to date. These and later SSFAs identified specific design bases issues that were entered into the appropriate corrective action system.

The Con Edison SSFA Program has contributed to improvements in overall system performance and related supporting activities. In the design area, it has contributed to improvements in Design Document control and quality, acquisition, or limited reconstitution of information and improvements in the plant modification process.

From the initial assessment, 87-11-A, the internal assessment process has proved to be a valuable tool in measuring the adequacy of the Con Edison processes in maintaining the design bases of the safety-related systems, beginning with the initial operation of IP 2. These assessments have sometimes identified unexpected results. Con Edison believes, however, that much has been learned from each assessment: not only that deficiencies may exist in a particular system but that generic trends may highlight problems in other systems. This has led to revisions and improvements in processes and procedures. The first assessment highlighted twenty-six Findings, thirteen observations, and eight discrepancies.

Con Edison believes that the knowledge gained by the early introduction and fairly comprehensive application of the internal SSFA process has enabled implementation of management measures to minimize identified problems. Con Edison believes that the quality of the internal SSFAs is as good as those that could be performed by an external party. The additional knowledge gained and the quality of the SSFAs conducted add support to Con Edison's confidence in the consistency between the design bases and the plant SSCs.

| SSFA No. | Safety System Functional Assessment |
|---|---|
| 1 | SSFA 87-11-A, Auxiliary Feedwater System |
| 2 | SSFA 87-11-C, Emergency Diesel Generators |
| 3 | SSFA 88-11-A, Residual Heat Removal System |
| 4 | SSFA 88-11-B, Liquid Radwaste System |
| 5 | SSFA 88-11-C, Electrical Power System |
| 6 | SSFA 88-11-D, Control Room Habitability |
| 7 | SSFA 89-11-A, Motor-Operated Valves |
| 8 | SSFA 89-11-B, Safety Injection System |
| 9 | SSFA 90-11-A, Instrument Air System |
| 10 | SSFA 90-11-B, Reactor Protection System |
| 11 | SSFA 90-11-C, Main Steam System |
| 12 | SSFA 91-11-A, Containment Cooling and Filtration System |
| 13 | SSFA 92-11-A, Alternate Safe Shutdown System |
| 14 | SSFA 92-11-B, Engineered Safeguards System Actuation |
| 15 | SSFA 93-11-A, Containment Spray System |
| 16 | SSFA 94-11-A, AFW/Electrical Ventilation/HVAC |
| 17 | SSFA 94-11-B, Cable Separation Program |
| 18 | SSFA 94-11-C, Motor-Operated Valve Program |
| 19 | SSFA 94-11-D, Service Water System Operational Performance Inspection (SWSOPI) |
| 20 | SSFA 95-12-11-A, CCR and TSC HVAC |
| 21 | SSFA 96-11-A, Plant Vent HVAC System |

Table 3.3.2.1, Indian Point Unit No. 2 SSFAs

3.3.2.2 Safety System Functional Inspection (SSFI) (NRC Initiated Assessments)

To date, two SSFIs have been conducted and followed-up at IP 2. The Component Cooling and Service Water Systems SSFI and the Electrical Distribution System Functional Inspection (EDSFI). In addition, the Service Water System Operational Performance Inspection was conducted like an SSFI.

Component Cooling and Service Water Systems Functional Inspection

The Safety System Functional Inspection of the Component Cooling Water System and the Service Water System (Inspection Report 50-247/88-200) was the first SSFI conducted by the NRC at IP 2. The inspections identified some strengths associated with Con Edison initiated functional assessments of the systems as well as a number of weaknesses which were subsequently resolved and closed.

EDSFI

The Electrical Distribution System Functional Inspection (EDSFI) was an NRC initiative to which IP 2 responded with a major design review and reconstitution effort over a three year period. This in-depth review of all aspects of the Electrical Distribution System significantly increased the knowledge of and confidence in this portion of the design bases. The EDSFI is described in detail in Section 4.1.3. It is listed here for completeness and to emphasize the contribution that this inspection and related activities have made to the overall confidence in the consistency between the design bases and the Electrical Distribution SSCs.

Service Water System Operational Performance Inspection (SWSOPI)

Although it was not formally an SSFI, the SWSOPI was conducted with NRC agreement, in accordance with an NRC Temporary Instruction (TI-2515/118) and with an NRC observer on site. Therefore, although it is not technically an SSFI, it is included here to better demonstrate the scope and depth of the SSFIs.

In October 1994, Con Edison received NRC approval for the IP 2 plan to conduct a Service Water System self-assessment. The inspection was conducted in accordance with the NRC's Temporary Instruction (TI) 2515/118, Service Water System Operational Performance Inspection (SWSOPI). The objectives of the self assessment as stated in the TI were to:

1. Assess the licensee's planned and completed actions in response to Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment;
2. Verify that the Service Water System is capable of fulfilling its thermal and hydraulic performance requirements and is operated consistent with its Design bases; and

3. Assess the Service Water System operational controls, maintenance, surveillance, and other testing and personnel training to ensure that the Service Water System is operated and maintained to perform its safety-related functions.

With NRC approval, Con Edison assembled an assessment team of three contractors and two Con Edison employees under the direction of a Con Edison team leader. The team also used a Con Edison employee as technical advisor. The Con Edison employees assigned to the assessment team were dedicated to this review and were relieved of their normal duties so that a thorough review of the Service Water System would be performed. The assessment team was mirrored by a Con Edison response team that was formed to expeditiously answer all issues raised by the assessment team.

The scope of the self-assessment was clearly defined before any onsite activities. The IP 2 Nuclear Quality Assurance (NQA) Department developed guidance that contained instructions on how to conduct the assessment, including the assignments for each team member. Tasks were broken down into areas of review corresponding to TI2515/118 requirements. These areas were then assigned to specific responsible team members for execution.

The assessment team was able to address all the inspection requirements of the TI in detail sufficient to assure themselves of adequate assessments in each assigned area. As a result of this review, the team generated 131 documented requests for information (RFI). Of these, forty were directly related to the design area. A number of these RFIs did require corrective actions or led to enhancements of the system. For example, the station has developed a new, upgraded, comprehensive flow model of the Service Water System to evaluate various station conditions as they arise. In the end, the team determined that the Service Water System had extensive thermal and hydraulic design margins.

Related to the SWSOPI results, the plant is installing newly designed SW pumps. The new pumps have design characteristics identical to the original pumps, but the physical and geometrical configuration has been upgraded to improve the pump reliability, availability, and maintainability. One pump has been installed, and the remaining five pumps are scheduled for installation in 1997.

Based on a comparison of the results of the SSFAs with the SSFIs, a strength of the SSFAs has been many instances where the assessments found the IP 2 programs for maintaining plant configuration consistent with the original design bases. Through a process of identifying weaknesses, as well as highlighting strengths, the SSFA program demonstrates that configuration control is continuously being maintained by a process of self-identification of problem areas and subsequent resolution. The SSFA program also demonstrates that plant physical and functional characteristics are being evaluated on an ongoing basis for consistency with the plant's design bases.

### 3.3.2.3 Self-Assessments

Department and section self-assessments are an integral part of the management processes at Con Edison. SAO-140, Indian Point Self-Assessment Program, is the procedure governing the self-assessment program. For purposes of the IP 2 Self-Assessment Program, a self-assessment is a department's or a section's critical review of its own performance or review of work processes or procedures (regardless of organizational boundaries) to determine whether:

- Performance requirements are adequate and consistent with objectives for excellence concerning
    1. plant and personnel safety,
    2. compliance with regulatory requirements and commitments, and
    3. plant reliability, efficiency, and competitiveness;
- Appropriate responsibilities have been assigned for the attainment of these requirements;
- Effective and efficient methods have been established for the attainment of these requirements; and
- The requirements are being consistently achieved in actual practice.

### 3.3.2.4 External Evaluations

External evaluations of Con Edison processes provide additional assurance that the plant processes are effective in maintaining the design bases and consistency between the Design Bases and the SSCs.

### 3.3.2.5 Nuclear Quality Assurance Department Audits

Sections 3.2.5 and 3.4.7 discuss the NQA Audit Program. NQA audits contribute to the assurance that SSCs are maintained consistent with the design bases by auditing operations, maintenance, testing, and modification, including jumper, activities for conformance with established procedures. SSFIs have provided an independent check of the quality of the internal audit program.

The results of all the SSFAs and SSFIs, together with the results of the internal audit program, do not identify a generic problem in the integration of the design bases of IP 2 with the operations, maintenance, and testing areas. Isolated problems may be identified in one audit, and upon resolution a later audit may cite this area as a strength. Plant practices over time have therefore proved successful in controlling the design bases in the procedures used by Maintenance, Operations, and Testing.

3.3.2.6 Independent Oversight

Section 3.4.5 contains a description of and the processes in place for the Station Nuclear Safety Committee and the Nuclear Facilities Safety Committee. By the nature of their oversite role, these committees provide additional confidence that the processes and procedures in place are functioning as intended and that consequently, the plant SSCs are being maintained consistent with the design bases.

3.3.3. Operations Support Processes

The processes discussed in this Section, verify in part that the activities performed, provide assurance that the plant SSCs are consistent with the design bases.

3.3.3.1 Operability Determinations

The process of ensuring operability for safety or safety support systems is ongoing and continuous. Section 3.4.4.1 describes this process and discusses how it fits in the overall corrective action process. One aspect of the process of determining equipment operability, when it is called into question, is an assessment of whether the SSC meets its design criteria with corrective actions and follow-up as required. This process helps to provide assurance that the SSCs are consistent with the design bases.

3.3.3.2 Walkdowns

Walkdowns conducted for specific programmatic activities have provided a measure of validation of SSCs consistency with the design bases.

System Walkdowns

System Engineering, in conjunction with other plant organizations, conducts documented walkdowns of select plant systems. These System Walkdowns are documented in System Walkdown/Status Reports. Their purpose is to record any deficiencies identified during the walkdown, along with the appropriate corrective action system entry identification numbers used to address the issues (i.e., Work Orders, Open Item Reports, Building and Grounds Request, or Request for Engineering Services). The report typically includes a section addressing the state of the system, which gives a general overview of the system, identifying unusual equipment limitations or concerns. On Maintenance Rule systems, the reports are to include an evaluation of system performance against the Maintenance Rule goals and performance criteria established to monitor plant maintenance and to ensure that systems will perform their safety functions when called on. The process by which these walkdowns take place is defined in System Engineering Procedure SE-Q-12.101, System Engineer/Specialist, as well as SAO-220, Plant Condition Inspection, and SAO-450, System Engineer/Specialist & System Performance Teams.

Although these walkdowns look at a wide variety of items, they focus on a number of design bases items as noted in Addendum 8.2 of SE-Q-12-101. Some examples of items that should be identified and corrected include:

- condition of snubbers, pipe supports, hangers, and fasteners;
- unauthorized modifications, partial modifications, or temporary modifications not on drawings;
- condition of barrier penetrations and seals; and
- condition of cable trays and barriers.

IPEEE Walkdowns

The Individual Plant Examination for External Events (IPEEE) included plant walkdowns to identify issues associated with each of the major external events examined. These efforts focused on beyond design basis events and the walkdowns did not require a verification against the design bases. Where issues were identified that could involve the plant design bases, those issues were submitted for resolution through an internal plant process.

The walkdowns were performed by teams including both Con Edison and contractor personnel. Walkdown plans were developed consistent with the focus of each external event. For the seismic portion of the IPEEE, the walkdowns were coordinated with those performed for the A-46 (SQUG) effort. The scope of the IPEEE walkdowns varied and included the following categories:

Winds:

- To identify structural features that might impact the ability of the structure to withstand pressure forces
- To identify potential susceptibilities to wind generated missiles
- To identify the potential for consequential failures of structures due to failures of other structures

Seismic:

- To identify structural features that might impact the ability of the structure or components to withstand various levels of seismic acceleration
- To examine component anchorages for obvious weaknesses (note that information gathered in the A-46 walkdowns was also used for IPEEE)

Fires:

- To identify electrical cabinet characteristics (sealing and venting)
- To identify the proximity of combustibles to ignition sources
- To identify the proximity of detectors to potential fire locations
- To determine separation distances within Central Control Room (CCR) cabinets

Floods:

- To identify ingress and egress areas
- To identify areas of influence for spraying or impingement
- To determine critical heights at which equipment could be affected

All the walkdowns were performed using walkdown plans. Interaction issues were also examined in these walkdowns (e.g., a seismic event that could induce fires).

Fuse Program

A self-initiated program was begun in the late 1980s to upgrade fuse information contained in the plant design documentation and to institute controls to assure the proper replacement and installation of fuses. In 1988 and 1989, a series of NQA surveillances were performed for the power fuses at all plant Class A Motor Control Centers. The data was used to assure the consistency of the as-installed condition versus the plant drawings. Additional data collection walkdowns were performed for CCR fuses during the 1991 and 1993 refueling outages. Numerous reviews and coordination studies were performed to assure the technical adequacy of the design. These reviews covered:

- 480 V feeds to MCCs
- Safety-related MCCs
- Non-safety-related MCCs
- 125V DC Power Panels
- CCR Supervisory and Flight Panel
- Miscellaneous panels

Based on these studies, enhancements were selectively made to the plant configuration. Additional efforts that are being undertaken include the failure analysis of failed fuses, use of only qualified fuses, reduction in the types of fuses used, and the development of additional guidelines for the design and installation of circuit protection devices. As the program continues, the applicable plant drawings will be further upgraded to provide comprehensive fuse information and a data base will be used to promote easy retrieval of data.

Cable Routing Assessment

In late 1987, the IP 2 NQA department conducted a number of self-initiated inspections of selected plant cable routing configurations as part of the cable separation program. A significant part of this assessment involved walkdowns that verified the configuration of cable routing and separation. This program is further described in Section 4.1.2. The walkdowns help to provide assurance that the cable configurations are consistent with the design bases.

3.3.3.3 Testing

Routine Testing

The surveillance testing program provides assurance that components and systems can perform their design bases functions. The surveillance requirements specified in the TS, as well as the tests and inspections specified in the American Society Mechanical Engineers (ASME) Section XI In-Service Testing (IST) Program, are performed to verify equipment and system performance against selected design parameters and functions based on design bases information.

Tests are conducted in accordance with approved test procedures to implement the surveillance requirements of the IP 2 TS, and to insure the return to service of equipment following maintenance.. These tests are administratively controlled by procedures. Guidance necessary to require that test procedures incorporate design bases information, as well as requiring that revisions to test procedures do not invalidate design bases conditions, has been proceduralized.

The ASME Section XI Pressure Test program for ASME Class 1, 2, and 3 components and systems provides verification that components and systems can perform their pressure integrity design functions. This program is governed by the IP 2 TS Section 4.2. Specification 4.2.1 directs that the IST pumps and valves be conducted in accordance with the applicable edition and addenda of Section XI of the ASME Boiler and Pressure Vessel Code as required by 10 CFR 50, Section 50.55a(g), which refers to Section 50.55a(f).

Con Edison submitted to the NRC its Third Ten-Year Interval IST Program on December 30, 1993, to cover the period from July 1, 1994 through June 30, 2004. On November 30, 1995, Con Edison submitted Revision 1 to its program submittal in response to questions included in NRC's November 30, 1994 Safety and Technical Evaluation Report. This program as submitted invokes the 1989 edition of Section XI of the ASME Boiler and Pressure Vessel Code. In the 1989 edition of the code, Subsections IWP and IWV, require that pump and valve testing be conducted in accordance with ASME/ANSI OM-1987, Operation and Maintenance of Nuclear Power Plants, Parts 6 and 10, respectively. TP-SQ-11.017, ASME Section XI - In-Service Test Program is the IP 2 Test and Performance Section administrative document that implements the requirements of the NRC approved IST Program submittal. Surveillance tests are written and approved to meet the requirements of the program.

When plant systems are modified, the modification package identifies what testing is necessary to demonstrate the as-modified equipment operable. In this way, the plant SSCs are assured as consistent with the design bases in a controlled manner when changes are made by the modification process.

Test Results

When surveillance procedures are completed, the results are evaluated for whether they are satisfactory or unsatisfactory. This determination is based on the test procedure acceptance criteria. If the test is determined to be unsatisfactory, a written Significant Occurrence Report (SOR) in accordance with SAO-124, Oral Reporting of Non-Emergency Events and Items of Interest and Significant Occurrence Reporting, is completed. The SOR is reviewed by the Daily Management Review Group under SAO-132. A priority assignment is established, that can result in a determination of cause, and development of corrective actions. These corrective actions are tracked through CITRS. If during the conduct of a test, conditions are found that do not otherwise fail the test but require correction, a work order is written to address the condition.

3.3.3.4 Operating Experience

Generic and operating plant information from the NRC and the industry is reviewed for applicability to IP 2 to determine whether actions are required to prevent conditions that occurred elsewhere and whether changes are needed to prevent inadvertent alteration of design bases or to initiate design changes. This operating experience program is described in more detail in Section 3.4.1.2. This program adds to the assurance that the design bases and SSCs are consistent because the significant issues raised at other plants and reported to the industry are evaluated for applicability to IP 2 and appropriately corrected.

3.3.3.5 Vendor Information

Key safety-related vendors are contacted annually to ensure that Con Edison has the latest vendor manual information in accordance with NRC Generic Letter (GL) 90-03 as required by SAO-409, Vendor Information Review Program. Vendor manuals are maintained by the Document Control Center, and their content is reviewed by system engineers.

3.3.3.6 Licensee Event Reports

Licensee Event Reports (LERs) are required to be submitted to the NRC by regulation and procedure for specific types of events. SAO-125, Station Written Report Requirements, Addendum III, defines the reportability criteria for LERs. Since 1973, when the plant was initially licensed, approximately 680 LERs have been submitted. A review of these LERs identified 23 LERs that discussed potential design issues in the description of the initiating event. Further review of these 23 LERs concluded that 16 of the events were related to issues involving the consistency of the plant configuration and the design bases. In none of these cases was the event considered of such significance that the plant was required to shutdown. Based on this history, Con Edison's confidence in the consistency of the plant SSCs with the design bases is reinforced.

### 3.3.4 Specific Initiatives

The verification nature of the activities performed, in the plant support processes, discussed in this section provide assurance that the plant SSCs are consistent with the design bases.

#### 3.3.4.1 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 10 CFR 50.65. The Maintenance Rule Program implementation included a number of activities that examined system functions and evaluated the station's historical performance.

The Maintenance Rule Scoping Committee reviewed plant SSCs to determine whether individual plant systems fit any of the following criteria as defined by NUMARC 93-01:

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

In addition to this review, each system engineer identified the functions performed by each system using both plant and industry data. Each of these functions was then evaluated against the above criteria for presentation to the station's Maintenance Rule Expert Panel.

Technical Specifications, UFSAR, DBDs, operating procedures and EOPs, Probabilistic Safety Assessment/Individual Plant Examination (PSA/IPE) reports, industry 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, subsystems, trains, subtrains, and groups of components supporting that function were included in the scope of the Maintenance Rule Program.

The Maintenance Rule Expert Panel provided feedback, comments, and support to the system engineers who evaluated which non-safety-related SSCs can prevent safety-related SSCs from fulfilling their intended function or can cause trips or safety system actuations.

The Con Edison Maintenance Rule Structural Monitoring Program establishes the procedures for satisfying the provisions of 10 CFR 50.65, The Maintenance Rule which deals with monitoring the conditions of buildings and structures and Con Edison's Maintenance Rule Program Plan. This program is intended to implement the guidance for complying with 10 CFR 50.65 provided in NUMARC 93-01, Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.

Performance Criteria

During the scoping process, the Maintenance Rule Expert Panel reviewed the risk significance of each system function, and each system engineer presented the performance criteria that would be used to evaluate each function. When possible, the performance criteria were fed back into the Station's PSA to evaluate the effect on core melt frequency. A three-year historical search of the corrective action systems (PPMIS, OIR, SOR, LER) was performed to demonstrate plant performance against these criteria. Where appropriate, "a)1" action plans were put in place to improve this performance. An ongoing monitoring program was also introduced to insure that plant performance is tracked against the established criteria.

3.3.4.1 Maintenance Rule Implementation

The Maintenance Rule Program implementation is described in SAO-160 was implemented in accordance with SAO-160, Maintenance Rule Implementation. Although this program requires the participation of many organizations in the plant, the System Engineering section is responsible for periodically monitoring the performance of the plant SCCs (for which specific criteria have been established) against those criteria. System Engineering walkdowns conducted in accordance with this program are described in Section 3.3.3.2.

Satisfying the intent of the structures monitoring program for those structures within the scope of the Maintenance Rule is accomplished by monitoring the condition of the structures through regularly scheduled inspections. Inspection checklists are used to record results. The frequency of inspections is given in the program. The inspection results are measured against various criteria including, but not limited to, original plant design bases, current use of structures, current codes and standards, and codes in effect at the time of construction. If degradation or deficiencies are found, the deficiencies must be evaluated to determine their category and any required action. All degradations and deficiencies must be documented and tracked using SORs or OIRs. The performance criteria for buildings and structures are based on a comparison between original or current design bases and a physical inspection or evaluation of existing conditions. The inspection results are categorized into one of three categories identified in the

Section, “Evaluation of Results.” The intent of the IP 2 Maintenance Rule Structural Monitoring Program is to have no Category A conditions for buildings and structures within the scope of the Maintenance Rule. If a Category A deficiency is encountered, the structure is considered to have failed the performance criteria.

In addition, if the condition of a structure is classified as unacceptable (Category A), the following actions are required:

a) Identify the event, situation, or condition causing the unacceptable condition,
b) Determine whether the degradation is a maintenance preventable functional failure (MPFF). If a structure has experienced a MPFF, a cause determination of appropriate depth must be performed and an action plan, to prevent the event from recurring, must be developed, and,
c) Determine whether a goal is required. If required, the goal and monitoring requirements are established.

Functional failures, MPFFs, and unavailability hours are used in various combinations to monitor system performance.

Changes to Design Basis Documents (DBD) are reviewed by System Engineers and Engineering Analysis. Design modifications are similarly reviewed by system engineers for changes to system functional scope and potential risk significance changes. EOP procedure changes impacting Maintenance Rule scoping are also reviewed. Any changes resulting from any of the above processes are addressed by system engineers.

3.3.4.2 Operating Equipment Program

The Operating Equipment Program was implemented primarily to verify that the as-built condition of the plant conforms to the plant design. Design related documents that were used as a basis for comparison to the field conditions included system diagrams, drawings, plant manuals, and equipment qualification (EQ) master lists. Additional information describing this program can be found in Section 4.2.2. Discrepancies between the plant as-built condition and the design related documents used for comparison were identified and prioritized for resolution. Ancillary activities in this project included field labeling of more than 6,000 components and the development of part lists for about 2,500 of the major plant components. The Operating Equipment Group maintains and enhances the equipment data base by reviewing plant modifications, resolving OIRs, incorporating equipment classifications, and updating equipment histories. This program is another example of activities that provide additional confidence in the consistency between the plant SSCs and the Design Bases.

3.3.4.3 UFSAR Review Program

This program is described in Section 3.1.4.2. It provides additional confidence in the consistency between the SSCs and the design bases.

3.3.4.4. Design Bases Document Initiative

The Design Bases Document Initiative is discussed in Section 4.2.3. The plant plans to continue this program to strengthen its usefulness as a source of design document information.

3.3.4.5 Other Engineering Programs

Sections 3.1.2, 3.1.3, 4.1 and 4.2 contain descriptions and discussions of a number of specific programs that Con Edison has undertaken to contribute to the understanding, documentation, and improvement of plant design documentation. In addition to the programs and modifications discussed in this report, the following list of modifications and programs undertaken to upgrade the plant SSCs and improve the plant processes and information have contributed significantly to the confidence level in the consistency between the plant configuration and the design bases:

Power Uprate Program
24 Month Fuel Cycles
Post TMI Modifications
Three Header Service Water System Program
Simulator Upgrade
Ultimate Heat Sink Study
Radiation Monitoring Upgrade
Emergency Diesel Engine Upgrade

These programs and the level of review they have received by Con Edison and by external organizations add support to the rationale that the design bases are understood and reflected in the plant SSCs.

3.4 Request

**(d) Processes for Identification of Problems and Implementation of Corrective Actions, Including Actions to Determine the Extent of Problems, Action to Prevent Recurrence, and Reporting to the NRC.**

Response

The breadth of the corrective action processes and the level of management review and independent oversight of its various aspects provide Con Edison with reasonable assurance that deviations from the design bases are identified and promptly reported to the NRC, are evaluated, and actions as appropriate are instituted to correct them and to prevent their recurrence.

Indian Point Unit No. 2's (IP 2) corrective action process is initiated by the identification of deficiencies or conditions potentially adverse to quality. SAO-113 and SAO-124 list numerous conditions adverse to quality which are required to be reported. A sufficient number of these conditions have a relatively low threshold for identification, reasonably assuring that potential design bases deviations will be identified. Once a potential problem is identified, it is entered into the appropriate tracking system, an initial evaluation of its significance is made, and operability or reportability determinations are conducted. An investigative priority is assigned, which defines root cause evaluation for more significant items. Development of proposed corrective actions, and consideration for potential generic applicability follows. The process is driven by the activities of the Daily Management Review Group (DMRG), plant senior level management, the Nuclear Quality Assurance (NQA) Department, the Station Nuclear Safety Committee (SNSC), and the Nuclear Facilities Safety Committee (NFSC), all of which perform various reviews of events and specific corrective actions. Senior plant management evaluates the overall corrective action program quarterly. Independently, the NQA Department periodically assesses the effectiveness of the program for tracking and implementing corrective actions intended to prevent event recurrence. Although seldomly occurring, issues identified relating to design bases are entered into appropriate corrective actions system for resolution. Should any similar design basis issues be identified in the future, they will be handled in the same appropriate manner.

3.4.1 Problem Identification

Station Administrative Orders (SAO) address the basic processes used at IP 2 for problem identification. These basic processes are:

(1) Significant Occurrence Reports (SORs): SAO-124 and SAO-132
(2) Open Item Reports (OIRs): SAO-113
(3) Work Orders: SAO-204
(4) Radiological Occurrence Reports (RORs): SAO-313
(5) Employee Concerns Program: SAO-123

(6) Compliance with Title 10, Part 21 of the Code of Federal Regulations: CI-250-2
(7) Operating Experience Review Program: SAO-420

Using these processes, plant staff is continually encouraged to identify known or potential problems and deficiencies to achieve a reasonably low threshold for problem identification.

3.4.1.1 In-plant Problem Identification

The DMRG, representative of plant middle management as outlined in SAO-132, reviews OIRs, RORs, and SORs, prepared since the previous DMRG meeting. The group assigns an investigative priority, initially determining what depth of evaluation must be performed, assigns action parties and initial due dates, reviews the results of completed actions or the status of evaluations yet to be completed, and judges the adequacy of the evaluations and the resultant approved corrective actions. The composition of DMRG helps to assure that potential conditions adverse to quality will be identified and objectively reviewed. Problems identified and actions to prevent recurrence are tracked to completion.

IP 2 has recently instituted the use of the Condition Identification and Tracking System (CITRS) which combines formally separate systems. CITRS tracks identified deficiencies or events, assignment of actions, action due dates, status of actions, and contains summaries of the results of evaluations and investigations. It is an enhanced management process for tracking and monitoring identified deficiencies through resolution. CITRS also provides enhanced capabilities for trending of information.

Periodic assessments of the status of corrective actions are performed by station departments and management. Additionally, audits are performed by the NQA Department. Trending reports are periodically published to enhance management's ability to assess the potential impact of trends and to assist in evaluating the effectiveness of corrective actions to prevent recurrence. A quarterly management review of the corrective action program and problem identification processes is required.

To ensure that personnel have appropriate mechanisms to identify problems, even anonymously if they choose, Con Edison has an Employee Concerns Program (SAO-123) which falls under the Corporate Code of Conduct for Employees (CP 100-1). SAO-123, Personnel Safety Concerns, was first effective January 2, 1981. This SAO was intended to describe employee options to express concerns about nuclear safety. The current Employee Concerns Program, SAO-123, was established at IP 2 in 1992. It has undergone several more enhancements and revisions in the ensuing years. A cornerstone of the IP 2 Employee Concerns Program is the establishment of an open environment which encourages identification of problems by providing a mechanism to report problems anonymously, if desired. Such concerns are reviewed by one of the Nuclear Ombudsmen. If the concern is substantiated, actions are identified and tracked to satisfactory

Edison is evaluating improvements to SAO-123. One of the proposed improvements will solicit all employee and contractor concerns, not just those limited to nuclear safety issues. Improvements to the Employee Concerns Program will strengthen problem identification in an already receptive environment.

A major effort started in 1996 to enhance departmental self-assessments. Effective response to issues identified during these self-assessments will be stressed to management and supervisory personnel. (See Section 3.3.2.3)

SAO-204, Work Order Procedure, establishes the administrative controls for initiating, approving, processing, implementing, and documenting work orders for maintenance activities. In this procedure, plant personnel and plant support personnel, who discover a deficiency with plant SSCs are to initiate a work order to correct the deficiency or report it to their supervisor. Once a deficiency is identified, SAO-204 controls the process, in conjunction with department-level procedures, to correct the problem. If an employee identifies a nuclear safety issue, or, under SAO-124, an impact on plant or equipment operability, the employee is to immediately notify the Senior Watch Supervisor (SWS). If the SWS (Operations Management) cannot determine whether the system, structure, or component is operable, the System Engineering Section and the Nuclear Safety and Licensing Department make an operability determination as described in Section 3.4.4.1.

System Engineering, in conjunction with other plant organizations, conducts walkdowns of various plant systems annually. The process by which these walkdowns are performed and documented is described in System Engineering procedures. The system walkdowns are documented in System Walkdown/Status Reports. These reports record any deficiencies identified during the walkdowns and the appropriate corrective action system entry identification numbers used to address the issues (i.e., Work Orders, Open Item Reports, Building and Grounds Request, or Request for Engineering Services). A more detailed description of walkdowns is contained in Section 3.3.3.2 of this report. Descriptions of the Open Item Report (OIR) and Significant Occurrence Report (SOR) problem identification processes are provided in Section 3.4.2.

Shortly after the promulgation of 10 CFR 21, Con Edison issued CI-250-2, Compliance with Title 10, Part 21 of the Code of Federal Regulations, to establish a uniform method of compliance with Part 21 by all departments involved with activities concerning nuclear systems and/or components. This Corporate Instruction has been revised several times in response to changes in the regulation or changes to the Con Edison corporate organization structure. CI-250-2 includes the procedure for identifying the nature of a condition that is adverse to quality and that may be reportable to the NRC in accordance with the guidelines of 10 CFR 21. This identification can be made by any employee or contractor employee. Steps outline transmitting such a condition identification through line supervision to the Manager of Nuclear

Safety and Licensing (NS&L), who is responsible for reviewing the concern to determine whether a detailed evaluation is appropriate. The NQA Department audits compliance with 10 CFR 21 and CI-250-2. Although some recommendations have been made, no major concerns with the program have been identified.

#### 3.4.1.2 Industry Operating Experience

Generic and operating plant information from the NRC and the industry are reviewed for applicability to IP 2, to determine what actions are required to prevent occurrence at IP 2 and to decide whether changes are needed in the plant design or licensing bases. Two procedures, CI-250-1, Correspondence To and From the NRC, and SAO-420, Operating Experience Review (OER) Program, govern handling of all generic information received by Con Edison. NRC Bulletins and Generic Letters requiring a response to the NRC receive mandatory processing under CI-250-1. Actions identified in developing the Con Edison responses, including those related to design bases, are tracked to resolution in CITRS.

SAO-420 governs the review and disposition of operating experience information not requiring a response to NRC. Incoming information requiring review is specified and assigned for evaluation. These reviews are tracked. If the review results in an implementing action, requirements are tracked to completion in CITRS. Actions could be necessary to maintain or enhance the design or licensing bases. If the action amounts to a change, e.g., modification or procedure revision, in the description of design bases, separate administrative controls (e.g., modification procedures) governing modifications or procedure revisions are invoked to make the requisite change. Vendor bulletins and information are additionally reviewed for incorporation into vendor manuals, under SAO-409, Vendor Information Review Program (See Section 3.3.3.5.)

### 3.4.2 Root Cause and Corrective Action Determinations

A description of the operability and reportability determination process is provided in Section 3.4.4. A description of the process for performing 10 CFR 50.59 Safety Evaluations is in Section 3.1.4.1.

#### 3.4.2.1 Root Cause Determination

The root cause analysis program at IP 2 is a multi-level process designed to provide the appropriate level of review for event or equipment problems. Currently, several SAOs can initiate the requirement to perform root cause analysis. These include: SAO-124, Oral Reporting of Non-Emergency Events and Items of Interest and Significant Occurrence Reporting; SAO-132, Analysis of Station Events and Conditions; SAO-113, Open Item Reports, Deficiency Reports and Stop Work Authority; SAO-160, Maintenance Rule Implementation; and SAO-313, Radiological Occurrence Reports.

SAO-124 identifies the requirement to document, in a Significant Occurrence Report (SOR), conditions such as plant operating anomalies that may include test failures, unexpected plant response, and removal from service of equipment identified in the TS. SAO-113 describes the process for addressing non-conformances, Deficiency Reports, and Stop Work Orders by the development and implementation of corrective actions to prevent recurrence. SAO-313 describes the process for identifying, reporting, and recording events classified as Radiological Occurrences which identify potential non-compliance radiological events. These items are reviewed under SAO-132 at the DMRG meetings to determine what type of evaluation, if any, is required. In this forum, these items are presented, discussed, and assigned one of three action levels. A level one item requires a fully detailed root cause investigation, formalized documentation, presentation to the Plant Manager, and assignment of corrective actions and may also require an LER. The next lower level requires that a review be performed of an event or failure, the root cause be determined and presented to DMRG, and corrective actions be assigned. Those items deemed to be least significant after initial DMRG review are still trended for historical purposes.

SAO-160 defines the responsibilities and overall program for implementing the provisions of 10 CFR 50.65. The program is intended to implement the guidance provided in NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plant(s) - Revision 1, dated January 1996. Part of the program calls for performing a cause determination for any repetitive functional failure of a SSC for any functional failure of a risk-significant SSC, or for the failure to meet any performance criterion or goal. This cause determination may include, as appropriate, formal root cause analysis.

3.4.2.2 Open Item Report Process as Part of Corrective Action Determination

The OIR program is a process for reporting non-conformance and conditions potentially adverse to quality and for identifying and implementing corrective actions. As appropriate to the event, root cause is considered and actions to prevent recurrence are implemented. To establish the lowest reasonable threshold for problem identification, any employee or contractor can submit an OIR. Examples of conditions when an OIR should be written include:

- Identification of various types of conditions adverse to quality, including:
    a. Potential Technical Specification violations
    b. Personnel safety violations
    c. Procedure violations
    d. Inadequate procedures or instructions
    e. Inoperability of a SSC
- A discrepancy between a controlled drawing and the actual field condition
- The discovery of foreign material in a system
- Defects in installed equipment
- Deficient conditions identified in the field such as:

    a. Loose supports
    b. Pressure boundary leaks
    c. Unexpected conditions identified during maintenance activities or plant tours
    d. Boric acid residue on components or structures
    e. Damaged wiring
- Inspections or activities not meeting acceptance criteria
- Tracking non-conformances, such as temporary repairs
- Inaccuracies found in records or documents
- Asking questions of either a technical or non-technical nature, particularly when the answer may result in identifying a condition adverse to quality

OIRs receive an initial review by the originator's supervisor, the NQA evaluator, or a Nuclear Ombudsman. As early in the process as feasible, if a condition that may impact operability is identified, the Operations Manager or the Senior Watch Supervisor (or both) is promptly notified. When a technical evaluation by Engineering is needed to assess operability, the item is immediately forwarded to Engineering, and an evaluation is performed. In most cases, this type of response is considered immediate, except as described in Section 3.4.4.1, below. The DMRG reviews the OIR on the next working day. The DMRG evaluates the OIR for potential impact on plant operations, including operability concerns and reportability. In addition, they discuss the item and may recommend actions which could include the need for performance of a root cause analysis. NQA participates in the DMRG and may include additional requirements from the perspective of the Quality Program. It also assigns actions to individuals as appropriate to resolve the matter.

Items are considered, as apparent, for impact on the UFSAR, Operating Equipment Data Base (configuration control), and potential effect on DBDs. When such impacts are identified, actions are assigned to the appropriate groups to review and provide corrective actions for the issue involved. When all actions deemed appropriate for an OIR are assigned and resolved, the action is reviewed for closure. The OIR is normally closed when the evidence is provided or referenced indicating that the required actions have been completed. Some OIRs are closed and tracked via other management processes, such as the Work Order process or the issuance of a Temporary Procedure Change (TPC).

The OIR and SOR processes described each have a flow path for corrective action review to ensure that the work performed has been satisfactorily implemented. This may include review by the originator, an assigned reviewer, or a knowledgeable assignee. The actions are documented in CITRS and then closed, when appropriate, following completion of the necessary actions. OIR and SOR status are reviewed quarterly by senior plant management. An annual review of OIRs is performed and periodic reviews of plant trips and human performance events are also performed.

3.4.3 Review of Deficiencies for Generic Applicability

The corrective action program is designed to correct deficient conditions and to evaluate them for potential generic issues. Each of the different programs contains these elements. The corrective actions recommended are reviewed and issued for implementation. These items are tracked and monitored in CITRS. Each of the individual sources, (SORs, OIRs, RORs, and Operating Experience Reviews) can also be individually monitored.

3.4.4 Operability Determination/Reportability Compliance

3.4.4.1 Operability Determination

The process of ensuring operability of safety or safety support systems is ongoing and continuous. Verification of operability is performed through surveillances and is supplemented by ongoing processes including check-off lists (COLs), walkdowns, NQA audits and design reviews. Formal determinations of operability are made when the normal verification processes result in a question of operability, the answer to which is not readily apparent from existing documented sources. Before April 1996, these determinations were routinely documented using the OIR process as described by SAO-113 or the SOR as described by SAO-124, though this was not a requirement. To enhance this process, in April 1996, IP 2 developed procedure SE-SQ-12.317, Equipment Operability Assessment, to evaluate plant systems' or components' capability of performing their safety or safety support functions in light of the identified anomaly. In developing this procedure, guidance was taken from the NRC Inspection Manual Part 9900-Technical Guidance, NUREG-1022, Licensee Event Report System, Description of System and Guidelines for Reporting - Revision 1, and NUREG-0580, Regulatory Licensing Status Summary Report, Nuclear Power Plants Data for Decisions. Under the SE-SQ-12.317 process, an equipment assessment determination documents the applicable TS or License commitments and evaluates how these items are potentially impacted by the identified deficiency. The allowable time to complete the operability determination is guided by TS limiting conditions for operation (LCO). In the absence of an LCO, the procedure nominally allows 24 hours. Any compensatory actions taken, as well as any follow up actions that might be required, are also identified. During the 1996 System Engineering Self-Assessment, the operability determination process was evaluated and determined to be a noteworthy strength.

3.4.4.2 Reportability Compliance

SAO-124, Oral Reporting of Non-Emergency Events and Items of Interest and Significant Occurrence Reporting, and SAO-125, Station Written Report Requirements, govern the Con Edison process for the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73, respectively. SAO-124 describes the process for making oral reports required by various regulatory agencies by Con Edison Corporate Policy and their documentation on the SOR form. These reports establish and maintain open lines of communication for reporting any change in status or items of interest at IP 2. The SAO identifies the responsibilities of various station personnel in making reportability determinations and notifications. SAO-125 delineates responsibilities for written reports to NRC and Other agencies.

### 3.4.5 Station Nuclear Safety Committee

The Station Nuclear Safety Committee (SNSC) is a standing committee of plant personnel established as required by Technical Specifications and in accordance with SAO-404, Station Nuclear Safety Committee. Its Chairman reports directly to the Vice President, Nuclear Power. The SNSC is comprised of the senior level plant staff required by the TS and also includes the additional membership of the Nuclear Quality Assurance Director, the Nuclear Safety and Licensing Manager, the Field Engineering Manger, the Generation Support Manager, and the Test and Performance Manager. This diversity in membership permits a multi-disciplined review of potential changes to the facility. One of the purposes of the SNSC is to provide an additional level of review of plant changes and procedure revisions, and their effects on nuclear safety. SNSC also performs plant post-trip evaluations and startup authorization reviews, reviews Licensee Event Reports (LERs) and reviews selected priority SAO-132 event reports. During the course of its deliberations, the SNSC determines whether additional corrective actions are warranted or if additional review is required. The SNSC can develop an issue-specific follow-up action or develop longer-term action items to be performed by assigned personnel. Thus, the SNSC provides an additional level of review of proposed corrective actions and of selected plant events for impacts to the safety of the plant.

### 3.4.6 Nuclear Facilities Safety Committee

The Nuclear Facilities Safety Committee (NFSC) is a standing committee that functions as required by Technical Specifications. The majority of its membership is required to be independent of the Nuclear Power organization. NFSC members are senior-level personnel who provide an independent review and audit of designated areas such as a reactor operations, radiological safety, electrical, mechanical and nuclear engineering, and administrative controls and quality assurance practices. They review safety evaluations completed under 10 CFR 50.59 for changes to procedures, equipment, or systems and tests or experiments to verify that they did not constitute an unreviewed safety question. They also review proposed changes to procedures, equipment, or systems or proposed tests or experiments that involve an unreviewed safety question under 10 CFR 50.59. Additionally, audits of facility activities are performed by the NQA Department with NFSC approval. One of the specific audits, required to be performed every six months, encompasses a review of the results of actions taken to correct deficiencies that may affect nuclear safety.

### 3.4.7 Formal Evaluations / Assessments

There are additional internal audits conducted by the Nuclear Quality Assurance Department, many of which are mandated by the IP 2 TS with the objective of confirming that activities are performed in accordance with the TS. Other internal audits concentrate on areas of importance, such as Fire Protection, Environmental Qualification and Seismic Qualification. All of these audits constitute a factual assessment of the activities audited, and approximately 35 audits of this nature are conducted annually. These audits also examine aspects of the corrective action process.

### 3.4.8 Vertical Slice Evaluations

As described in Section 3.3.2.1, 21 SSFAs have been performed by Con Edison to date, using the general methodology that the NRC developed for performing Safety System Functional Inspections (SSFIs). The issues from the self-initiated SSFAs have been reviewed, as well as the NRC's assessment of Con Edison practices reflected in the two SSFIs conducted by the NRC. The issues requiring further action have been entered into CITRS and are being tracked to resolution.

### 3.4.9 Self-Assessments

A major effort was begun in 1996 to enhance Nuclear Power departmental self-assessments. These enhanced self-assessments included reviews of Operations, Maintenance, I&C, Radiation Protection, and System Engineering activities. As part of this continuing enhancement effort, the need to properly and expeditiously address issues identified during these self-assessments is stressed to management and supervisory personnel.

3.5 Request

**(e) The Overall Effectiveness or Your Current Processes and Programs in Concluding That the Configuration of Your Plant(s) Is Consistent with the Design Bases.**

Response

Current work processes, plant programs, and the effective implementation of associated procedures described in this response, provide Con Edison with reasonable assurance that the configuration of Indian Point Unit No. 2 is consistent with the design bases. The processes utilized for controlling design and plant configuration information are continuing to evolve and will improve the level of accuracy and accessibility of such information. The implementation of the corrective action program and processes utilized for problem identification; determination and implementation of corrective actions which prevent recurrence; and operability and reportability determinations to the NRC, provide confidence that a significant condition adverse to quality, when identified, is appropriately addressed.

Periodic assessments of the status of corrective actions are performed by station departments, management, and the Nuclear Quality Assurance Department. Trending reports are periodically published to enhance management's ability to assess the potential impact of trends and to assist in the determination of the effectiveness of the corrective actions to prevent event recurrence. A quarterly management review of the corrective action program and problem identification processes is also performed. Comprehensive management overview, coupled with independent oversight by the Nuclear Facilities Safety Committee (NFSC) and the Nuclear Quality Assurance Department (NQA), help to ensure that the plant activities and programs described in this report are properly implemented. The effectiveness of these programs has led Con Edison to conclude, with reasonable assurance, that the configuration of IP 2 is consistent with the design bases.

## 4.0 Design Review and Reconstitution Programs

**Indicate whether you have undertaken any design review or reconstitution programs. If design review or reconstitution programs have been completed or are being conducted, provide a description of the review programs including identification of the systems, structures, and components (SSCs). The description should include how the program ensures the correctness and accessibility of the design bases information for your plant and that the design bases remain current.**

Response

The cumulative impact of the activities and programs performed by Con Edison to date, i.e., design review and limited reconstitution programs, plant upgrades, design changes, and system assessments, has improved the level of detail and breadth of IP 2 design bases information. Although Con Edison does not have a formally designated reconstitution program, the plant's design bases have been and continue to be reconstituted on a case by case basis, as needed. Many of the programs described in this section have resulted in some design bases information enhancements.

### 4.1 Major Design Bases Upgrade Programs

The following list represents a sampling of programs, design modifications, and evaluations where design bases information was significantly upgraded or reconstituted. This list is not all inclusive but is presented to indicate the magnitude of the past and current Con Edison efforts. Some major efforts include:

1) Motor-Operated Valves (MOV) Program (Generic Letter 89-10)
2) Cable Separation Program
3) Electrical Distribution System Functional Inspection (EDSFI) Program Elements
4) Bus Loading Program
5) 480 Volt Bus - Degraded Voltage Program

#### 4.1.1 MOV Program

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

This multi-year activity, verified that all safety-related MOVs, via calculation and test, were capable of performing their design functions under worst-case conditions.

As part of the program, design requirements, such as differential pressure (delta-p), flow, process temperature and ambient environmental factors,) related to MOV operation and applicable information contained in the UFSAR, Technical Specifications, Design Bases Documents (DBDs), System Descriptions, and Operating and Test procedures were reviewed to establish the hydraulic and environmental conditions under which a specific 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. These calculations were prepared under the guidance of the former Central Engineering-Project Engineering Nuclear department (now Nuclear Power Engineering) and were reviewed and accepted by the applicable system engineers and the Nuclear Safety & Licensing (NS&L) Department. These calculations are accessible through the IP 2 Calculation Indexing System and have been provided to the IP 2 DBD group.

After the design requirements were established, each MOV assembly was reviewed to assure that it was capable of functioning under design conditions. Specifically, minimum available electric power, motor torque, and gearing were evaluated to assure that each MOV would operate under required conditions. Minimum, maximum, and target thrust windows were established consistent with the particular valve and actuator requirements and limitations. Motor protection requirements (thermal overloads) were evaluated and weak link evaluations were made. Specific calculations were made for voltage drop from the Motor Control Center (MCC) to the MOV, weak link, thermal overload, and MOV capability. Each evaluation was documented in calculational form and reviewed and approved as required by OP-290-1, Engineering Operations Manual, Section 5.

As a result of these reviews, marginal MOVs (just meeting the minimum standards) were identified and modifications developed and implemented to enhance those margins. Modifications included actuator upsizing, motor upsizing, gearing changes, power cable upgrades, spring pack changes, and thermal overload relay changes. Modification packages were developed under the IP 2 modification program as specified in OP-290-1 (See Section 3.1.1.2) using the generic modification format. A standard design evaluation was developed and each proposed modification to a specific MOV was evaluated. For example, upsizing an actuator would add weight and possibly change an MOV's center of gravity, this change could negatively impact the seismic capability of the installation. Therefore, for an actuator upsize, the design criteria required that the pipe/pipe support system be evaluated for the change in weight, or center of gravity. Generic safety evaluations meeting the requirements of 10 CFR 50.59 were prepared, reviewed, and approved for all proposed modifications.

In addition to the activities described above, a test program was prepared to perform design bases differential pressure (delta-p) and flow testing on each MOV, where practical, to validate the calculational methodology used to verify MOV capability. Static and dynamic testing of safety-related MOVs were performed during the 1993 and 1995 refueling outages. Test results correlated well with design information for the majority of valves tested. In a few instances, valve factors were higher than anticipated. However, the actuator was capable of providing the torque/ thrust associated with these higher valve factors so that MOV operability was maintained. Dynamic test conditions ranged from 10% - 100% of design bases differential pressure with approximately half of the valves tested at or near full design bases differential pressure.

Con Edison continues to participate in, and monitor industry MOV activities, including EPRI's MOV Performance Prediction Program Users Group, the MOV Users Group, NEI, Westinghouse Owners Group (WOG), and others. Industry information is channeled to the MOV system engineer and NPE via the Operating Experience Review Program. Con Edison is currently participating with the WOG to developing positions for long term maintenance and testing of MOV performance in response to Generic Letter 96-05.

4.1.2 Cable Separation Program

The Cable Separation Program was performed to establish formal design criteria regarding cable separation requirements. Once the design criteria were established, field verifications and resultant design modifications were performed to ensure the plant configuration in fact met these design criteria. This overall effort significantly enhanced Con Edison's understanding of the plant's design bases regarding electrical train assignments, raceway physical layout, and application of cable separation criteria.

Several self-initiated inspections of selected plant cable routing configurations were conducted in late 1987 by the NQA Department. Anomalies found at that time were documented in two OIRs. These issues were resolved and a plan was developed to expand the area of field walkdowns and engineering evaluations. As the walkdowns and engineering evaluations were conducted and broadened in scope through 1988-89, it was determined that the multi-organizational effort needed to be formalized in a more structured program. The program document was issued in November 1989 with the purpose of validating and reconstituting the design bases for the Cable Separation Program at IP 2. Other goals of the project included updating the plant raceway schematics, routing drawings and cable schedules; and installing field upgrades in selected cases.

Walkdowns were completed in all Class A areas of the plant, except for those scheduled on selected trays and cables in the Cable Spreading Room. Over 6,000 cables have been walked down for the following voltage categories: heavy power, medium power, small power, and control and instrumentation.

This program has produced the following results:

- assurance that the evaluated field conditions meet the applicable single failure criteria,
- assurance that the plant design documentation reflects field conditions,
- reconstitution of the Cable Separation DBD.

Reports have been written on an area-by-area basis documenting the status of the walkdowns, engineering evaluations, and corrective actions such as the installation of approximately 250 blankets and sheet metal barriers and the installation of 25 additional fuses to enhance the protection of critical circuits that have been completed. Dedicated sections in each report addressed, where applicable, specific design bases findings deduced from a comprehensive review of the as-installed cable routing configurations. The results were documented in the Cable Separation DBD issued in June 1993.

The improved understanding of the design bases resulting from this program has been communicated to engineers, and designers in a design criteria document and in a series of training courses. This improved understanding has enhanced Con Edison's ability to maintain the plant's configuration. In addition, various procedural controls have been incorporated into the design change process to assure that cable separation considerations are adequately addressed through the design and installation phases of plant modifications.

4.1.3 Electrical Distribution System Functional Inspection Program Elements

The Electrical Distribution System Functional Inspection (EDSFI) was an NRC initiative to which IP 2 responded with a major design review and reconstitution effort over a three-year period. It involved over seven man-years of effort for the initial preparation and inspection not including major parallel efforts for the Cable Separation and EDG Upgrade Programs. Activities included test program development and implementation, engineering evaluations, field walkdowns, procedure revisions, and the performance of transient loading calculations. In addition to the internal review effort, issues were identified through participation in the nationwide EDS Clearinghouse, which accumulated questions and experience from other EDSFIs, and direct participation and support of the New York Power Authority in their preparation and EDSFI for Indian Point 3. An improved understanding of the plant's design bases resulted from the NRC EDSFI and the Con Edison activities. The Electrical Distribution System was demonstrated to be capable of performing its design functions.

The NRC Team reviewed the results of three previous NRC Special Inspections and the resolutions of those findings, as well as currently available calculations, design documents, and test data. The scope of the NRC Inspection included: 345 kV and 138 kV offsite power grids, gas turbines and 13.8 kV bus, unit auxiliary, station auxiliary and station service transformers, 6.9 kV system, emergency diesel generators (EDGs), 480 V safety-related unit substations and motor control centers (MCCs), station batteries, battery chargers, invertors, 125 VDC safety- related buses, and the 120 VAC vital distribution system.

Issues were identified and tracked in a specially developed EDSFI Action Item Tracking System. Issues with potentially adverse safety impacts were entered into routine Corrective Action Systems and reporting mechanisms. Items were appropriately closed out as noted in NCTS and subsequent NRC correspondence and inspection reports. During the EDSFI inspection, the NRC reviewed: AC and DC systems loading; voltage regulation during normal and degraded grid conditions; sequencing of engineered safeguards equipment onto the preferred power supply and EDG; short circuit protection, including overload protective devices, for AC and DC electrical equipment; ratings of EDS equipment; and protection of Electrical Containment Penetrations. The heating, ventilating and air conditioning (HVAC) systems that ensure an adequate operating environment for the safety-related equipment in the Diesel Generator Building, the Switchgear Room, the Cable Spreading Room, and the Battery Rooms were also reviewed. Additionally, walkdowns of the fuel storage and transfer system, EDG starting air system, lube oil and jacket water systems, and Service Water System were conducted.

The NRC Team reviewed both procedures and guidelines governing the EDS design calculations, design control, plant modifications, and power demands of major loads and the translation of mechanical into electrical loads used as input into the design bases calculations. The maintenance and test programs developed for plant modifications were reviewed to determine their technical adequacy.

In many cases, the Con Edison long-term responses to EDS issues have gone substantially beyond mere compliance and have incorporated enhancements to the plant's design including:

- Double breaker installation for Battery 21/22 bus tie
- Seismic Evaluations for all 480 Volt breaker rack out configurations, as early as 1991
- Modifications to EDG transfer switches & field flash circuitry.
- EDG Building 6th fan installation
- Degraded Voltage Studies Set point Changes and Condensate Pump trip
- Agastat relay replacements with Tempo relays (Scheduled for 1997 RFO)

### 4.1.4 Bus Loading Program

Con Edison performed a comprehensive evaluation of the 480 Volt buses to ensure that the EDS was capable of supplying connected loads under various operating configurations including plant startup, normal operation, and loss-of-coolant accident (LOCA) with offsite power. The results of the evaluations confirmed that overload conditions do not exist under these configurations and that the EDS was capable of meeting the plant's design bases.

The 6.9kV system supplies the four 480 Volt buses (2A, 3A, 5A and 6A) via four 6.9kV/480 Volt service transformers. The 125 VDC system consists of station batteries 21 through 24, and associated panels. The 118 VAC instrument bus system consists of Invertors 21 through 24, and associated instrument buses.

Con Edison contracted Ebasco to perform a loading study of the 480 V buses and to develop and to provide a load tracking program. This program lists both the connected and operating load for each load component and sums the operating loads to arrive at the total bus loads. Additionally, to ensure that the 125 VDC system has the capacity to supply the required 125 VDC loads during a two hour loss of AC, loads are also tracked by using a spread sheet program developed by Ebasco. The Ebasco spreadsheet program calculates the load on each bus and produces a text and graphical output of load vs time. Load profiles are then developed from this data. Battery sizing calculations for Batteries 21, 22, 23, and 24 were released in September 1989 and were updated following the 1995 refueling outage.

The Ebasco 480 V Load Bus study was released as a calculation in January, 1991. The 480 V bus loading was assessed in September, 1991 and May, 1993. The 480 V bus study had been updated during the 1995 refueling outage. To ensure that the instrument bus system, including the inverters, has the capacity to supply the required 118 VAC load, instrument bus loads were also tracked and updated following the 1995 refueling outage.

During the 1991 EDSFI (50-247/91-81), the NRC Team reviewed the battery sizing calculations and the battery loading calculations and found that the batteries had sufficient spare capacity for the anticipated two hour loading duration. In addition, the team reviewed the battery charger calculations and concluded that the battery chargers could supply the maximum 125 VDC system loads and simultaneously recharge two partially discharged batteries within 15 hours as stated in the UFSAR. The NRC Team also considered the use of the Indian Point Unit No. 1 batteries for non-safety related loads as a good initiative to increase the capacity and availability of the IP 2 batteries.

4.1.5 480 Volt Bus - Degraded Voltage Study

Following the 1976 Millstone event, the NRC requested that utilities install additional protective circuits to protect against degraded voltage conditions on the safety related buses. A number of studies and upgrades have been performed by Con Edison in this area. In 1981, new undervoltage relays were added and a Technical Specification change was approved by the NRC. In 1984, circuitry was modified to accelerate the transfer from offsite power to the EDGs during a degraded voltage condition coincident with a SI signal.

Prior to the 1991 NRC EDSFI the degraded voltage study was updated. Enhancements were made during the 1993 and 1995 refueling outages to preclude any unnecessary transfer from preferred power to emergency onsite power during normal plant operations, unit trip or accident conditions. New degraded voltage setpoints were submitted and approved by the NRC in 1993 and implemented during the 1995 refueling outage.

## 4.2 Other Upgrade/Review Programs

A number of major programs, projects, and upgrades have been, and continue to be, conducted by Con Edison. These are characterized as review programs, rather than reconstitution programs and are described in Sections 4.2.1 through 4.2.9. The following is a summary.

Accuracy and availability of design information has been improved with review of modification packages, operating equipment, and design bases documentation. A review of modification packages was completed during 1989 and 1990 that enhanced the modification files. An Operating Equipment Project was initiated in 1987 to verify that the as-built field installation of IP 2 conforms to the design documents. Data was obtained for 40,000 plant components for the operating equipment database. The Design Bases Review Project began as an effort to capture the original design documentation for IP 2 and is currently being broadened in scope to include additional design information and to improve accessibility of the information.

In 1988, Con Edison performed an EDG Load Study to obtain flexibility for future plant changes. This study was completed in 1990 and is routinely updated after refueling outages. In 1991, Con Edison completed a diesel upgrade program.

Several types of plant modifications have been made and will continue to be made on a case-by-case basis. Protection device coordination studies were performed for selected electrical systems to maintain the integrity of the electrical systems for specific initiating events. Modifications and industry information are reviewed to maintain the integrity of the electrical power system. HVAC systems have been upgraded on an as-needed basis. Electrical plant upgrades have been made to add features not present in the original plant design. These improve the plant capability to provide safety functions. The design of Service Water Pumps is being upgraded to improve reliability, availability, and maintenance of the pumps.

### 4.2.1 Reviews of Modification Packages prior to 1989 Including Calculations & Safety Evaluations

In 1989 and 1990 reviews of previously installed IP 2 modification packages and files took place. The purpose of the initiative was to enhance the early IP 2 modification packages and project files by making them more complete and provide easier access.

The contractor, Burns & Roe, input approximately 1,300 modifications into a modification tracking system. The contractor reviewed the modifications and files to determine if the required records were contained in the packages. A retrieval effort was undertaken to collect the vital information identified as missing from the individual modification files. As a result, these IP 2 project files are now more complete and the information is more centralized and readily retrievable. The modification packages and associated project files are a source of information for the DBDs and design documents.

### 4.2.2 Operating Equipment Project

The Operating Equipment Project was implemented primarily to verify that the as-built field installation conforms to the design documents. The project was self-initiated in late 1987 and involved obtaining data for over 40,000 plant components via plant walkdowns using contractor inspectors. The project was governed by a plan for an Operating Equipment Database for IP 2, and was supplemented by activities addressed during the walkdown phase including: component data development and verification, component parts lists development, database management and integrity, and discrepancy resolution.

Discrepancies between the as-built condition and the design related documents were identified and prioritized for resolution. A high priority for resolution (priority 1) was assigned to those cases where components were in the field but not shown on the drawings and those cases where the components continued to be shown on drawings but were not actually installed in the field. Of 20,000 discrepancies identified, nearly 5,000 were priority 1. Discrepancies were resolved primarily by field engineers and system engineers. By late 1993, all priority 1 discrepancies were resolved, principally by performing technical reviews to revise drawings to reflect the field conditions, or by changing the field to reflect the drawings.

Procedures have been written for control of changes to the operating equipment database as a result of modifications, maintenance, or updating of component data. The Operating Equipment group maintains and enhances the equipment data base by reviewing plant modifications, resolving OIRs, incorporating equipment classification, and updating equipment histories.

### 4.2.3 Design Bases Documents Review Project

The Design Bases Document Review Project was initiated in 1987 in an effort to capture the original design documentation for the plant and make it more accessible. To date, 22 safety significant systems have been investigated and Design Bases Documents (DBDs) have been prepared with their sequence of completion based on probabilistic risk assessment (PRA) studies.

The 22 DBDs issued as of January 1997 include:

Fire Protection/Alternate Safeguards Shutdown System
125 Volt DC System Batteries and Distribution System DBD
Auxiliary Feedwater System DBD
Feedwater Steam Generator Water Level Control System
Containment Cooling and Filtration System DBD
Nuclear HVAC System
Component Cooling Water System DBD
480 Volt Electrical System DBD
Containment Isolation and Support System

Nuclear Instrumentation System DBD
Main Steam System DBD
Containment Spray System DBD
Overall Unit Protection System DBD
Chemical and Volume Control System DBD
Reactor Coolant System/Steam Generators System
Electrical Separation DBD
Residual Heat Removal/Safety Injection System DBD
Service Water System DBD
Emergency Diesel Generator DBD
Reactor Protection System DBD
Engineered Safeguards Systems System Actuation
Seismic, Structures and Devices DBD

During the development of a DBD, specific open items may be discovered. Criteria for determining the open items to be documented for resolution or tracking as punch list items are:

a. information that is incorrect or missing in controlled documents;
b. information that because of its use or importance should be controlled, but is not presently controlled;
c. information requiring confirmation (information is available from uncontrolled sources such as meeting minutes or unofficial correspondence);
d. information conflicting with other information, i.e., assumes both sources of information are controlled or controlled information differs from as-built configuration or licensing commitments.

The Program includes, initial safety screening of all discrepancies, prioritized based on criteria documented with DBD procedure 15.106, and tracking of open punch list items to final closeout via incorporation, as appropriate, into the next revision of the associated DBD.

The DBD's format and content are controlled by Technical Services Procedures DB15.101 to 110. Procedure DB15.103, DBD Writers Guide, describes how the document is formatted and structured. The DBD Writers Guide also states what kind of information is to be included and how the document is reviewed prior to issue.

The DBD program has included benchmarking at other utilities and, a user feedback questionnaire that has resulted in format changes to increase their ease of use. Further efforts are being made to this area including consideration of the use of improved media and technology to further improve accessibility and maintainability of the DBDs.

In the summer of 1996 the three year effort by the Westinghouse Owners Group and Westinghouse to identify, screen and scan over 6,800 design documents directly related to the design of IP 2 ended. An optical disk containing the design documents is now loaded on the IP 2 network. The files can be searched via the Nuclear Power Information Network (NPIN) computer system which provides the user the ability to view the images in the Nuclear Records Management Center (NRMC) using the FileNet software.

This DBD Project is continuing to expand in scope including performing additional DBDs selected by considering risk significance. This broader focus is to be coupled with lessons learned in assembling a more user-friendly tool and maintaining it up-to-date with more timely retrievability, content, and compatibility with the UFSAR and plant hardware and software evolutions.

All of the programs described above added significantly to the understanding of the IP 2 Design Bases. Con Edison continues to improve the various programs and further enhance the availability of design bases information.

4.2.4. EDG Loading Program

In 1988 Con Edison recognized the need to reevaluate (EDG) loading for providing flexibility in future plant design changes and contracted Westinghouse to perform such an analysis. The EDG Load Study was completed in July 1990.

Under OP-290-1, Engineering Procedures for Operating Nuclear Plants, the IP 2 engineer is required to develop design criteria for any modification. If there are any changes to the electrical load, the engineer is required to complete the Electrical Load Sheet and the Electrical Load Modification Guideline Sheets, which include the proposed change in EDG loading. By procedure, these changes are included in the EDG load-on-timing to verify EDG capability. When a modification is completed, this load is included as an installed load in an update of the EDG load study after each RFO.

4.2.5 EDG Upgrade Program

In December 1989, Con Edison performed a modification to achieve a higher rating for the Emergency Diesel Generators. This was accomplished during the 1991 Refueling Outage. Elements of the program included detailed evaluations by the original equipment manufacturers for the generator and diesel engine. Westinghouse was assigned to manage the program and Con Edison was responsible for performing required modifications to supporting systems. The program added significantly to design documentation by the performance of activities such as:

- Generator and ALCO Diesel Engine evaluation
- Full Scale Mock up Testing of Switchgear and Bus Duct

- Diesel Generator Excitation System Testing
- Acceptance Testing for EDGs
- ALCO Diesel Commercial Grade Dedication Program
- Documentation of EDG operating limits
- EDG Building Ventilation upgrade

With the flexibility provided by the upgrade, major improvements were made to rearrange 480 volt loads, improve separation and diversity, install new 480 volt MCCs, and achieve one step load management for loads requiring manual reconnection to the EDG.

### 4.2.6 Protective Devices Coordination Study

Protective device coordination studies have been performed for selected electrical systems. Protective settings and coordination criteria were prepared for new plant modifications and reviews of existing design. Areas of improvement were identified and appropriate modifications were implemented to refine protective devices coordination. Reviews are performed on industry-related problems for applicability to the Indian Point 2 Electrical Distribution System. IP 2 incidents are also evaluated to determine proper electrical responses for the specific initiating events. New modifications are reviewed for short circuit and protective devices coordination to maintain the integrity of the electrical power system. Surveillance testing and calibration checks are performed regularly for the protective devices and relays for the 345 kV, 138 kV, 6.9 kV, and 480 V electrical systems to assure that set points are within specification.

### 4.2.7 HVAC Upgrading Programs

The heating, ventilating and air-conditioning (HVAC) systems are upgraded on an as-needed basis, and the EDSFI, and system engineers' requirements. Key modifications include:

- Improving Central Control Room (CCR) Carbon/HEPA filter efficiency
- Rearranging the power feed to the Primary Auxiliary Building fans
- Technical Support Center HVAC installation to meet NUREG-0696
- Replacement of CRDM ventilation fans to improve cooling of CRDM gripper coils
- Separation of Switchgear Room fans power feed to separate MCCs

### 4.2.8 Electrical Upgrades and Improvements

Electrical plant upgrades have added capabilities and features not existing in the original plant design. Examples of these electrical plant improvements included:

- Adding of 3rd and 4th Instrument Bus Inverters
- Adding of associated 3rd and 4th Batteries
- Upgrading four instrument bus inverters to 10 kVA (original 7.5 kVA)

- Upgrading seven DC Auto-Transfer Switches
- Upgrading 480V Breakers w/Solid State Trip Devices
- Unitizing Batteries 23 and 24 for 480V Breaker and EDG DC Control
- Upgrading Battery 22 to 1800 A.H.
- Replacing DC Power Panel 21 and 22 breakers with higher short circuit rating breakers
- Adding 2nd DC Tie Breaker between Panels 21 and 22
- Adding four new Class 1E MCCs and one-step Load Management
- Unitizing 3rd Component Cooling Pump (modified power supply)
- Changing Inverter 23 alternate supply to prevent loss of a EDG from impacting two Instrument Buses
- Installing 6th EDG Bldg. Fan for loss of two EDGs
- Installing new CONAX electrical penetrations
- EDG Starting Air System upgrade to improve air start reliability
- Main Generator Replacement

Electrical system improvements include:

- reconnection of the Main Generator Emergency Bearing Oil Pump onto the IP 1 battery reduced safety-grade battery duty,

- use of 58 (versus 60) cell batteries reduced required charging boost voltage and associated duty on safety related loads,

- the continuously and frequently energized BFD relay replacement program was instituted, coupled with the relay Preventive Maintenance program and pre-baking program for NBFD relays to improve their reliability,

- the capacitor replacement program improved the life and reliability of Foxboro instrumentation modules. The NUS/ Haliburton Foxboro module equivalent replacement program has allowed an extended life of Foxboro process instrumentation system without a major change that would make it software and/or firmware dependent, and

- the Boric Acid Heat Tracing System has been improved over the original design.

The above examples and records of continuing plant improvements in electrical systems demonstrate Con Edison's continuing commitment to maintaining and improving the quality of plant equipment. These upgrades improve the plant's capability to support required safety functions which are included in the design bases.

### 4.2.9 Service Water Pump Upgrade

An in-depth scale model study of the service water pumps and bay was performed by Alden Research Laboratory, Inc. to obtain a thorough understanding of their characteristics. The in-depth study identified subsurface vortexing, silt deposition, and foreign material ingestion. The result of this study have been used in the Service Water Pump upgrade. New service water pumps were designed to improve performance and reliability.

Design improvements include:

- Bearing composition change from cutless rubber to Thordon (Thordon has proven to outlast cutless rubber by 10 to 1);
- Improved shaft coupling design to sleeve and clamshell configuration;
- Addition of turning elbow at under deck pump discharge to reduce internal losses;
- Metallurgy of all wetted parts to be non-corrosive stainless steel or Nitronics (this material enhancement will result in improved impact resistance by the introduction of tougher steels);
- Introduction of spacer coupling so that either packing or mechanical seal can be selected;
- Pump bowl assembly featuring key and thrust ring impeller construction.