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ComEd

February 6, 1997

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

Subject: Byron Station Units 1 and 2
Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information
NRC Docket Numbers: 50-454 and 50-455

References:
(a) J. M. Taylor letter to J. J. O'Connor dated October 9, 1996, "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Basis Information"

(b) T. J. Maiman letter to A. B. Beach dated November 12, 1996, "Programs to Improve the Quality, Maintenance and Accessibility of the Design Bases at ComEd Nuclear Stations"

(c) T. J. Maiman letter to A. B. Beach dated January 30, 1997, "ComEd Plan for Upgrading the Quality and Access to Design Information at All Six Nuclear Stations"

This letter transmits Byron Station's response to the Nuclear Regulatory Commission's (NRC) request for information under 10 CFR 50.54(f) (Reference (a)). For the reasons described in detail in the attachment to this letter, Byron Station concludes that there is reasonable assurance that its procedures accurately reflect its design bases and that it is configured and operated in a manner that is consistent with its design bases, as defined in 10 CFR 50.2, or as otherwise permitted under the NRC's regulations.

Byron's process for developing this response was structured to take a comprehensive look at the configuration management program as it applies to the design bases, and to assure accuracy and completeness. Verifications and reviews of the response were conducted at several levels, including reviews by site management, a ComEd Corporate Team, and Byron's Onsite Review. Finally, an external review team comprised of individuals who have extensive experience with the nuclear regulatory

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process and are not involved in ComEd's day-to-day activities provided an independent assessment of the quality, completeness and responsiveness of the reply.

These processes for preparing and reviewing the response provide a high level of assurance of the completeness and accuracy of the response.

The response is structured around the five action items in the 50.54(f) request. The attachment to this letter is supplemented with three appendices.

- Appendix I, "ComEd Organization Restructuring to Improve Byron Station's Ownership and Control of the Design Bases," discusses the establishment of the ComEd design engineering organization. This appendix also discusses supporting roles of other Corporate and Site Groups which oversee conformance with the design bases.

- Appendix II, "Configuration Control Processes," presents a summary of the major, common processes upon which each of the six ComEd Nuclear Stations have based their programs. This appendix supports Action (a) directly.

- Appendix III, "Nuclear Fuel Services' Design Processes," discusses the role of the Corporate Nuclear Fuels Group in supporting our six nuclear stations in reload analysis and fuel management.

This response is intended to provide detailed information on the action items in the 50.54(f) request. Descriptions of the programs and processes in the response are current processes as defined by existing station procedures. The programs and processes are subject to change as they are upgraded and enhanced per station procedures. This response captures and condenses a substantial body of information of the current programs and processes. Additional detail is available in other correspondence and company documents. Specific commitments related to the programs and processes described herein are contained in other relevant docketed correspondence. In addition, in order to alleviate any ambiguity as to ComEd's commitment to future actions regarding the quality, maintenance, and accessibility of design bases information, we have provided those commitments under separate cover to the NRC (References (b) and (c)).

Current Situation

Byron's conclusion that there is adequate assurance that it is configured and operated consistent with its design bases is based on several factors:

The plant was determined by the NRC to meet its design bases when it was licensed. ComEd certified that Byron had been designed, constructed, and pre-operationally tested in a way which would assure consistency with the FSAR, the NRC's Safety

Evaluation Report, and the Commission's regulations. Byron and Braidwood were constructed under the same procedures and practices and licensed under the replicate plant option of 10 CFR 50, Appendix N. The product of the combined inspection and assessment efforts on Byron and Braidwood provided a high level of assurance that Byron's initial configuration and performance were consistent with the design bases. Since then, changes to the plant's physical configuration and operating procedures have been made in accordance with the programs that were designed and adopted to assure continuing consistency with the design bases. Under those programs, significant changes to the plant's configuration and its operating procedures are subject to multiple reviews. These programs have been improved and upgraded over time. Byron has an accredited training program for all critical processes.

Years of normal operating experience have shown that the plant's structures, systems and components generally have operated as designed. A comprehensive program of inspection and surveillance testing is performed on an ongoing basis. Byron Units 1 and 2 have consistently responded as expected during unplanned events and transients.

Self-assessments, ComEd Quality Assurance audits, NRC inspections, and third party reviews have repeatedly probed the processes implemented to maintain procedures and structures, systems, and components consistent with the design bases. For the most part, these reviews have corroborated that the plant's procedures accurately translate design bases information and that its structure, systems, and components are consistent with their design bases. Where substantial discrepancies have been identified, their root causes and extent of occurrence have been determined, the discrepancies have been corrected, and the procedures that permitted them to occur have been strengthened by eliminating their underlying root causes. This experience leads Byron Station to conclude that its configuration management processes, as supported by its programs for operational monitoring and walkdowns, inspection, and surveillance testing, and corrective action, provide reasonable assurance that the procedures accurately reflect the design bases and that the structures, systems, and components are consistent with their design bases.

Byron and Braidwood are plants of relatively recent vintage. They were licensed in the 1980's in accordance with the Standard Review Plan and share an 18 volume UFSAR prepared in accordance with Regulatory Guide 1.70, Rev. 2. As a result, information regarding the design bases, as defined in 10 CFR 50.2, is contained in the UFSAR to a high level of detail and completeness. As part of the initial licensing process, ComEd certified that the plants have been designed and constructed consistent with the design bases information contained in the FSAR.

As described in Appendix I and other sections of the response, Byron and Braidwood have acquired the majority of the supporting design information and the calculations performed by the NSSS supplier and the architect engineer in the original design of the plants (and in subsequent modifications.) These calculations are generally complete,

indexed, and retrievable. In addition, select personnel who participated directly in the original design and construction of the plants are now part of the design engineering organizations currently in place at Byron and Braidwood. For these reasons, Byron and Braidwood have not found it necessary to collect the information already available into summary level Design Bases Documents (DBDs), or to undertake large-scale design review and reconstitution programs. However, Byron and Braidwood have undertaken specific initiatives which have reconstituted or established more specific calculations that implement the design bases where necessary, such as in the development of the motor-operated valve (MOV) program.

Future Action

Despite the substantial bases for concluding that Byron is configured and operated consistent with their design bases, recent inspections has heightened ComEd's concerns in these areas. Following recent inspections at Zion Engineering and Technical Support (E&TS), LaSalle Service Water Operational Performance Inspection (SWOPI), and Dresden Independent Safety Inspection (ISI), Mr. Thomas J. Maiman (Chief Nuclear Officer) communicated by letter dated November 12, 1996, to Mr. A. Bill Beach (NRC - Region III) (Reference (b)), that all six ComEd Nuclear Station would implement actions that would enhance ComEd's ability to operate and maintain its nuclear stations consistent with their design bases. These actions can be summarized as follows:

1) Establish an Engineering Assurance Group until normal engineering activities have improved;

2) Provide direction in procedures for resolution of design bases discrepancies;

3) Expand SQV audits of major contractors;

4) Define and reconstitute, as required, calculations that are critical to maintaining design control.

In conclusion, ComEd is dedicated to the safe operation of its nuclear power plants. We clearly recognize the importance of operating and maintaining Byron Station in conformance with design bases. The commitments in the Reference (b) and (c) letters, and the ongoing oversight roles by Site Engineering Assurance and Quality Verification groups, all contribute to enhance the current reasonable assurance that the stations will be operated and maintained within their design bases.

Please contact us should you have any questions on the attached information.

Very truly yours,

Thomas J. Maiman
Executive Vice President
Chief Nuclear Officer

Kenneth L. Graesser
Site Vice President
Byron Nuclear Power Station

TJM/tb

Attachment

cc: A. B. Beach, Regional Administrator - RIII
J. Callan, Executive Director for Operations
S. Collins, Director - NRR
G. Dick, Byron Project Manager - NRR
S. Burgess, Senior Resident Inspector Byron
Office of Nuclear Facility Safety - IDNS

COUNTY OF DuPage
STATE OF Illinois

AFFIDAVIT

I, Thomas J. Maiman being first duly sworn, do hereby state and affirm that I am the Chief Nuclear Officer for Commonwealth Edison Company, that I am authorized to submit the attached letter and attachments on behalf of the company, and that the statements in the letter and attachments are true and correct to the best of my information, knowledge, and belief.

Thomas J. Maiman

Thomas J. Maiman
Executive Vice President
Chief Nuclear Officer

Subscribed and sworn before me on this 6th day of February, 1997.

My commission expires 5-19-98.

"OFFICIAL SEAL"
MICHELLE KOWALKOWSKI
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 5/19/98

Michelle Kowalkowski
Notary Public

EXECUTIVE SUMMARY

The following is a summary of the corresponding sections of Byron Station's response to the NRC's October 9, 1996 request for information pursuant to 10CFR 50.54(f) regarding the adequacy and availability of design bases information.

Action (a): We have reviewed the Byron Station and related corporate engineering design and configuration control processes, including those that implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR Part 50. The processes are described in Section 1.0 and Appendix II of the response. While some programmatic deficiencies in processes and procedures have been identified, corrective actions have been implemented to address and prevent recurrence of these deficiencies. Overall, the results of the review of this section support our conclusion that the scope and extent of these processes are adequate to maintain the plant configuration and operation consistent with the design bases.

Action (b): We have made an assessment of the rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures. Byron's procedures are developed and implemented in a manner consistent with industry standards. Formalized checks and balances in the procedure preparation and revision processes and the required qualifications of the procedure reviewers provide assurance that the procedures capture applicable design bases requirements. Experience with the procedures has shown that actual results were as expected in responding to normal and abnormal events and when procedures were used for routine plant activities. The results of this assessment provide the bases for concluding that applicable design bases requirements have been translated into appropriate station procedures.

Action (c): Our assessment also addressed the rationale for concluding that structure, system, and component configuration and performance are consistent with the design bases. The formal certification process at the time of Byron's licensing provides the assurance that Byron had been designed, constructed, and preoperationally tested in accordance with applicable design requirements. Ongoing verification programs and audits of the work processes at Byron ensure that design bases are maintained while performing routine activities. Accordingly, there is reasonable assurance that system, structure, and component configurations are consistent with the design bases.

Action (d): We have reviewed the processes for identification of problems and implementation of corrective actions at Byron, including actions to determine the extent of the problems and actions to prevent recurrence. The processes have generally been found to be effective in identifying problems. Problems identified as significant are investigated to identify the fundamental cause(s) of the problems. The Site Quality Verification function has been independently assessed and found to be effective, broad based, independent, and probing. Problem reporting to the NRC has been found to be appropriate. The results of our review support our conclusion that the scope and extent of these processes is adequate and that design base issues are resolved in a timely manner.

Action (e): We have conducted an assessment of the overall effectiveness of the current processes and programs in concluding that the configuration of Byron Station is consistent with the design bases. These processes have evolved over time and continue to evolve as evidenced by the various improvement initiatives. Each component of the configuration control process has been reviewed and evaluated via internal audits, NRC inspections and third party activities. Where deficiencies were found, the extent of their impact was determined and comprehensive corrective actions were taken to resolve problems and prevent their recurrence. The results of the reviews and assessments performed in connection with the preparation of this response provide reasonable assurance that the processes and programs are effective.

As described in the response, Byron and Braidwood have acquired the majority of the original design information and calculations performed by the architect/engineer and the NSSS supplier. This information is generally complete, indexed, and retrievable. Specific initiatives have been undertaken to reconstitute or establish specific calculations, such as in the development of the motor-operated valve program.

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1.0 ACTION (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

1.1 Introduction

ComEd's processes for engineering design and configuration control, including those that implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR Part 50, are described in this section. The requirements of ComEd's Quality Assurance Manual are reflected throughout these processes. These processes implement ComEd's configuration management model. It is followed at both the corporate office and the stations.

In the corporate office, implementation of the configuration management model as discussed in Appendix I is the responsibility of the Chief Engineer, Configuration Management, who reports directly to the Engineering Vice President. At the sites, implementation of the configuration management model is the responsibility of the Site Engineering Manager.

The complementary configuration management roles of the corporate office and site Engineering Departments are reflected in their implementation of ComEd's configuration management model. The corporate office is responsible for Nuclear Engineering Procedures (NEP) and Nuclear Station Work Procedures (NSWP). The sites are responsible for Administrative Procedures and NSWPs that implement the processes which constitute the configuration management model. The role of each is summarized below.

NEPs and NSWPs provide guidance on corporate expectations for configuration control processes to the stations. Their major elements relevant to engineering design and configuration control are summarized in Appendix II. Appendix II summarizes the major common processes used at all ComEd Nuclear Stations, and are implemented by station administrative procedures. A matrix in that appendix illustrates how the various processes relate to the configuration management model. The matrix also summarizes the processes for implementing 10 CFR 50.59 and 10 CFR 50.71(e).

The Nuclear Fuels Design Process is corporate sponsored for all stations and is discussed in Appendix III.

Station Administrative Procedures provide the details for implementing the elements of the configuration management model. These procedures specify how work is to be performed and how the station is to be operated to assure consistency with the design bases. Specific process differences or enhancements made at Byron Station which deviate from corporate procedures,

due to Byron Station needs, are discussed in Sections 1.4 through 1.11. As with all procedures, procedural adherence is a clearly communicated management expectation and processes are in place to reinforce that adherence.

Byron Station Administrative Procedures for engineering design and configuration control are structured to achieve the following objectives:

- Assure the establishment of adequate design controls that implement the quality assurance requirements in Appendix B to 10 CFR Part 50 as applied to new designs and design changes.

- Assure that changes continue to satisfy design bases requirements through controlled processes for review and approval of the design, its installation, its testing and its operation.

- Assure safety evaluations are in compliance with 10 CFR 50.59.

- Assure implementation of the Updated Final Safety Analysis Report (UFSAR) update requirements of 10 CFR 50.71(e).

- Assure that Quality Control (QC) inspections and post-modification tests are conducted for modifications.

- Assure the update of documents, databases and drawings that are affected by changes.

- Assure that field changes to a modification are subject to engineering approval.

- Enable procedure preparers and reviewers to have ready access to design bases information and are familiar with the design bases.

- Assure that personnel are trained.

1.2 Requirements of Procedures Which Control Design Bases

Byron station's procedures for implementing the configuration management model address four principal areas: (1) design control to determine the impact of proposed actions on consistency with the design bases; (2) configuration control to assure that documentation is updated after a change is made; (3) safety evaluation under 10 CFR 50.59 to determine if the change involves an unreviewed safety question (USQ) or Technical Specification Change; and (4) licensing basis review to update the UFSAR. The following overviews of the station's actions in each of these four areas provide a comprehensive summary of the important steps that are taken to maintain the plant's configuration and operation consistent with its design bases.

1.2.1 Design control procedures and processes conform to Criterion III, V, and VI of 10 CFR 50, Appendix B and include the following provisions:

- The procedures of the design control processes implement the requirements of the ComEd Quality Assurance Manual. They apply to new design work and design changes for safety-related and certain non-safety-related structures, systems, and components (SSC).
- New design work (including design changes) is reviewed for conformance with design bases (or appropriate changes are implemented in the licensing basis).
- New design work (including design changes) is documented in calculations, analyses, specifications, drawings, or other controlled documents.
- New design work (including design changes) is subject to design verification.
- New design work (including design changes) is approved by management.

1.2.2 Configuration control procedures include the following provisions:

- Prior to approval, design changes are required to be evaluated for conformance with design bases.
- Changes are verified to be consistent with the design bases.
- Prior to approval, design changes are required to be evaluated to determine their impact upon operating, maintenance, and testing procedures and training programs, and appropriate changes are required to be made to affected procedures and programs.
- Approved design changes are required to be implemented in accordance with controlled documents (e.g., work packages, installation procedures or specifications).
- Modifications are required to be subject to QC inspections and post modification tests.
- Changes in the field (e.g., temporary alterations) are required to be evaluated and are subject to engineering approval, where appropriate.
- Changes to operating, maintenance, and testing procedures are required to be reviewed to determine their conformance with design bases and other design documents.
- Changes in the plant are required to be reviewed under 10 CFR 50.59 to determine whether an unreviewed safety question exists or a Technical Specification Change is required.
- Responsible personnel are required to receive training in the above procedure(s).

1.2.3 Procedures for implementing 10 CFR 50.59 include the following provisions:

- Changes to the UFSAR, changes to design and operation, procedure changes, temporary alterations, and prolonged operation with degraded and nonconforming conditions.
- Changes are required to be screened to determine whether they involve an unreviewed safety question or Technical Specification Change.
- Technical changes in the UFSAR are required to have a documented 10 CFR 50.59 safety evaluation by the station or Safety Evaluation Report (SER) from the Nuclear Regulatory Commission (NRC).
- Safety evaluations are subject to review by qualified personnel.

- Unreviewed safety questions and changes to the Technical Specifications are required to be submitted to the NRC for approval.
- Responsible personnel are required to receive training in the 10 CFR 50.59 procedure.

1.2.4 Procedures for implementing 10 CFR 50.71(e) include the following provisions:

- The effects of safety analyses for license amendments are required to be incorporated in UFSAR updates.
- The effects of other safety analyses required to be submitted to the NRC are required to be incorporated in UFSAR updates.
- The updates to the UFSAR include not only changed information but also new information.
- Between updates, identified changes for the UFSAR are controlled and accessible to plant personnel.
- Responsible personnel are required to receive training in the above procedures.
- ComEd has an approved change to the submittal frequency as stated in 10 CFR 50.71(e). Updates are provided on a 24 month basis.

1.3 Overview of Processes Which Control Design Bases

Processes which control plant configuration are grouped according to their primary objective:

- Work Initiation
- Work Planning and Design
- Interim or Temporary Actions
- Work Execution
- Design Document Update

Flowchart A of Appendix II summarizes the relationships of the Engineering Design Control Processes and Configuration Control Processes.

Each of these objectives is discussed below.

1.3.1 Work Initiation

Work may be initiated via a number of processes. The Action Request (AR) is generally used for maintenance work with the exception of on-the-spot maintenance that is implemented by station procedure. For engineering assistance and evaluation, the Engineering Request (ER) is used. For problem investigation and corrective action, the Problem Identification Form (PIF) (discussed under Action d) is used.

Consistency of operation with the design bases is initially assured via the prompt screening of ARs, ERs, and PIFs. Each shift, the ARs and PIFs are reviewed by the Shift Manager or SRO

Licensed Designee for impacts on operability and Technical Specifications. The Shift Manager or SRO Licensed Designee determines whether the identified problem results in a safety concern and whether immediate action is required. If immediate action is required, the appropriate operations manager and the engineering department are notified. If a piece of equipment, important to safety, has been identified as being degraded, an operability assessment is performed by engineering. The initial determinations are reviewed regularly by an experienced, multidisciplinary screening committee, which includes representatives from Operations, Engineering, and Maintenance. The committee assigns a work priority consistent with the safety significance of the request and regulatory compliance concerns.

1.3.2 Work Planning and Design

Work is planned and work packages are prepared using the work control process (Section 1.4). Some work may require a plant modification, procedure change or new procedure, use of other than like-for-like replacement parts, or setpoint change. Design change (Section 1.5), procedure control (Section 2.0), parts replacement (Section 1.5.6), or setpoint change (Section 1.5.4) processes are then used.

Discrepancies between the plant configuration and the design documents will be brought to the attention of the engineering department. In this case, an appropriate request form is used to resolve document/plant discrepancies.

Replacement parts are evaluated by Parts and Materials Management personnel under a like-for-like replacement process in accordance with station procedures. If the work analyst identifies that a like-for-like is not available, an ER will be initiated for a non-like-for-like replacement.

Consistency between work and the plant's design bases is assured by the development of a work package that requires the consideration of design bases information, application of the materials and parts procurement process, and the incorporation of post-modification testing developed through either the Engineering Modification or Work Package development processes.

1.3.3 Interim or Temporary Actions

At times it is necessary to take interim action to correct a potential or actual condition adverse to quality, pending the completion of permanent corrective action. In such cases, operability determinations (Section 1.11) are performed to assess whether a SSC is capable of performing its specified function in its present condition and what, if any, compensatory action is required. Safety evaluations (Section 1.6) may be required depending on safety significance and time required to implement the long term corrective action. Work activity affecting the power block portion of the station undergoes an appropriate review per 10 CFR 50.59. Temporary Alterations (Section 1.5.3) are used to document the acceptability of an interim change to the plant configuration. Consistency of operation with the design bases pending completion of work is therefore assured by performing evaluations and/or taking compensatory actions and/or documenting the temporary condition as a temporary alteration.

1.3.4 Work Execution

Routine maintenance or a design change is executed with work packages prepared by maintenance analysts. Routine maintenance is implemented in the field by station procedure. Design changes are implemented primarily by NSWPs.

The Out-of-Service/Return-to-Service (OOS/RTS) (Section 1.8) process ensures that operational plant configuration is controlled consistent with the design bases during performance of maintenance activities. Work is controlled by station procedure or a work package prepared under the work control process. If changes are required to the original modification package, those changes are documented on a field change request and reviewed by engineering. Post-maintenance/modification testing ensures that the work is acceptable and that the equipment can be returned to service. Consistency between operation and the plant's design bases is maintained by the Work Control Process. If a special test is required it is prepared per station procedure.

Technical Specification surveillances, which verify design requirements, are also controlled through the work control process. This work is performed utilizing controlled procedures. Acceptance criteria failures are reported on a Problem Identification Form.

1.3.5 Design Document Updates

Design document updates may be required either because of a discrepancy, such as a deviation between the document and the as-built plant or because of a design change. The Document Change Request (DCR) (Section 1.10) process is used to control design document changes. Other update processes include UFSAR update (Section 1.7), and station procedure revisions (Section 2.0). Vendor manual and/or operating or maintenance procedures may need to be changed based on information received from the vendor as part of the Vendor Equipment Technical Information Program (VETIP) (Section 1.9).

Configuration control, accessibility, and retrievability of design bases information and change documents has been enhanced through the use of the Electronic Work Control System (EWCS). Information in EWCS includes revisions pending against all design documents.

Consistency between the plant configuration and plant documentation is assured by the document change and update processes.

1.4 Work Control Process

The work control process at Byron Station is designed to allow the plant to be operated and maintained consistent with the design bases. A combination of station and corporate procedures are in place to control the work process. Most work processed through the work control process is initiated by an AR except for on-the-spot maintenance. ARs are used to resolve documented problems with structures, systems, and components in the plant. A majority of the ARs are

initiated by Operating personnel but anybody in the plant has the capability and responsibility to initiate an AR. The initiation of ARs, as most work at Byron, is processed using the Electronic Work Control System. (See Appendix II, Process 15).

Operators may identify problems with plant equipment while making rounds in the plant. These problems are discussed with the Shift Manager and ARs are initiated to resolve these problems. A determination is made by the Shift Manager or SRO Licensed Designee if the identified problem results in a safety concern and if immediate action is required. If immediate action is required, the appropriate operations manager and the engineering department are notified. If a piece of equipment that is important to safety has been identified as being degraded, an operability assessment is performed by engineering.

Action Requests are reviewed by a screening committee. The AR Screening Committee makeup and responsibilities are discussed in Appendix II, Process 1. The responsibility of this committee is to prioritize and assign work and to identify design bases issues.

Once the committee has prioritized and assigned the AR to the proper group for resolution, the work package is initiated. The maintenance departments have procedures to control how they perform work. Work identified by the committee as affecting the design bases is processed and coordinated by a work analyst, and an engineering request is initiated. The resolution of these ERs by engineering could result in issuing a modification or an exempt change, a temporary alteration, setpoint/scaling request, etc. or a clarification response. The work analyst takes this information and uses it in the preparation of the work package.

Replacement parts are evaluated by Parts and Materials Management personnel under a like-for-like replacement process in accordance with station procedures. If the work analyst identifies that a like-for-like part is not available, an ER is issued for a non-like-for-like replacement part.

Post-Maintenance Testing (PMT) verifies that the equipment is performing as expected and that operability requirements are met after the work has been completed. Operations will not release a work package as being complete until all of the specified testing is complete. PMT requirements are incorporated into the work package instructions through the work control process. Failures of PMTs require a PIF to be initiated.

Safety-related work packages are reviewed to assure all work, including testing, is complete. Reviews are performed by various groups including, at a minimum, Systems Engineering, Operations, and Quality Control. A maintenance supervisor identifies the appropriate reviewers based on the work request type and the safety-related function. Special reviewers can be included for areas such as Inservice Inspection (ISI), Inservice Testing (IST), Equipment Qualification (EQ), etc.

1.5 Design Change Process

The Design Change Process at Byron is accomplished through the activities described in Appendix II. The following design change processes are discussed:

- Plant Modifications
- Exempt Changes
- Temporary Alterations
- Setpoint Changes, Electrical Trip Settings, and Overload Heater Size Changes
- Computer Software Revisions
- Technical Evaluations (Parts and Material Replacement)

1.5.1 Plant Modifications

A Plant Modification is a planned change in plant design or operation and is subject to design control measures commensurate with those applied to the original design. The current process for accomplishing a plant modification starts with the initiation of an ER. An ER can be requested by anyone on site, however most originate from ARs or are generated within the engineering group. After initial approval and assignments, a Modification Scope Meeting is held. This meeting provides an introduction of the modification to affected departments other than engineering and provides a vehicle for input from these groups. Required deliverables and preliminary affected documents are defined during the meeting. Designs for all plant modifications are issued via the Engineering Change Notice (ECN) process, (see Appendix II, Process 12). The design is then processed through a series of steps that include scoping activities, field walkdowns, preparing Design Input Requirements (DIRs), engineering calculations (see Appendix II, Process 17), and 10 CFR 50.59 screenings/safety evaluations. The DIR defines the major technical objectives, constraints, and regulatory requirements that govern the development of the design. Once the design change package is completed, technical and on-site reviews are initiated. After the reviews have been completed and the design package is approved, it is issued for work instruction preparation. Modifications are implemented in the field through the Modification Work Control Process.

In all cases, the design and engineering activities described in these processes are implemented at Byron by individuals who have been trained and are qualified to perform these functions. Although there are areas within the process that provide overall reviews of the design, specific areas provide for independent reviews against the design bases. The first area is the Engineering Change Notice which is used to develop the design. Each ECN is subject to an interfacing review process, an independent review, and an approval. Walkdowns performed after installation also provide another area where the design is evaluated to ensure that it has met the original design intent. If the modification is found to be installed outside of tolerance, or an alternate design configuration is required, a Field Change Request (FCR) is generated to evaluate the differences. All FCRs are subject to the same reviews and approvals as the original design. Additional

engineering calculations and 10 CFR 50.59 review may be required. Post-Modification Testing is the last area where the design is functionally evaluated to ensure it has met the design requirements.

1.5.2 Exempt Changes

Exempt Changes are design changes that maintain form, fit, and function, and require minimal engineering effort. Exempt changes cannot be used for any change that is determined to involve an unreviewed safety question, a change to the technical specifications, or a functional change. Exempt changes are approved through the same cycle as plant modifications. The mechanics of the exempt changes are similar to the plant modification changes as described above except for the modification scope meeting. Engineering Change Notices are required for all safety-related, augmented quality-related, and seismic designs.

1.5.3 Temporary Alterations

Temporary Alterations (Temp Alt) are alterations to the approved design configuration of a structure, system, or component that satisfy established criteria. The Temp Alt process is intended to ensure that a Temp Alt made to plant equipment does not degrade plant safety/reliability or unacceptably alter the approved design configuration. The engineer for a Temp Alt assembles the package by marking up copies of the affected design documents, performing a 10 CFR 50.59 screening/safety evaluation and preparing other pertinent documentation. Temp Alts are independently reviewed by a qualified engineer and the on-site review group. Critical Control Room Drawings (CCRD) affected by the approved temporary alterations are required to be updated.

Engineering is responsible for establishing and maintaining a controlled log and a documentation reference file of the Temp Alts reviewed by the engineering organization. Engineering is also responsible for a semi-annual review of all temporary alterations to determine whether the continued installation of the Temp Alt is technically acceptable, and that suitable progress is being made to permanently disposition the temporary configuration.

The Shift Manager or SRO Licensed Designee is responsible for initiating and reviewing the installation of Temp Alts. Temp Alts are usually installed using a work request (WR) and independently verified to ensure the changes are correctly installed. The Shift Manager or SRO Licensed Designee is also responsible for ensuring post installation testing identified by engineering is complete.

Temporary Alteration removal is authorized by the Shift Manager or SRO Licensed Designee. The removal is performed by using the WR process and includes independent verification of the restoration. Post removal testing is performed as specified by engineering to ensure the configuration has been properly restored and critical control room drawings are updated as required. (See Appendix II, Process 6)

1.5.4 Setpoint Change Control Process

The Setpoint/Scaling Change process controls the alteration of the existing design or function of a system or component by increasing, decreasing, or removing an existing setpoint. The process is controlled by a Station Administrative Procedure and is implemented by Site Engineering. The Setpoint/Scaling Change control program ensures that changes to setpoint values or scaling changes are initiated, analyzed, controlled, and documented in an approved manner. Examples of components to which the Setpoint/Scaling Change control program applies are the following:

- Process Control Instrumentation
- Alarms, Annunciators, and Monitors
- Electrical trips, Interlocks and Permissives
- Heater Sizing
- Relief and Safety Valves

Setpoint change requests are initiated by completing a Setpoint/Scaling Change Request (SSCR) form and by initiating an Engineering Request (ER). Proposed changes to setpoints are reviewed for consistency with all aspects of the plant's design bases. Changes to affected procedures must be made concurrent with the installation of the new setpoint or scaling. Cognizant station departments are responsible for updating the affected Operating, System Engineering, and Maintenance procedures immediately prior to installation of the SSCR. The review process ensures that changes are reviewed by appropriate departments. The Setpoint/Scaling Change Document Revision Checklist identifies all procedures, control room critical drawings, design documents, drawings, etc., that must be updated due to the pending SSCR. The SSCR training requirement checklist identifies required training that must be accomplished prior to implementing the change. A 10 CFR 50.59 Screening/Safety Evaluation must be performed and processed as part of the package. The engineering supervisor determines the disciplines required for the On-Site review of the SSCR package. After review and approval by the on-site review group, the Station Manager signs for final approval of the package. Implementation of the setpoint change is by the use of a work request and the work control process. (See Appendix II, Process 9)

1.5.5 Computer Software Revisions

The Byron Computer Software revision program applies to software that is safety-related, used to perform controlled work, used to verify Station Technical Specifications compliance, or used to comply with the regulatory requirements. This process specifically describes the program to control revisions to engineering software. Once a need to develop or revise Engineering Software has been identified, a Software Activity Request is generated to describe the circumstances and identify the activities that need to be performed. Once the request has been reviewed and approved by the Site Software Administrator, a Software Management Plan (SMP) is generated by the software owner or designee. The SMP includes identification of the software product, responsibilities and schedules, required documentation, required reviews and other similar technical and administrative items. After approval of the SMP by the Site Software Administrator, the Software Requirement Specification (SRS) is developed. The programming

changes will then begin based on the documents generated above, in preparation for software testing. A preliminary test case is used to validate the Engineering Computer Program to assure that the software produces correct results for the test case. (See Appendix II, Process 11)

1.5.6 Technical Evaluations - Parts and Material Replacement Process

Parts and replacement materials are procured through a centralized material procurement process. It establishes uniform criteria for procurement of safety-related items and services that will be used for operations, maintenance, and modification of ComEd nuclear stations and includes the following objectives:

- Ensure installed items are suitable for the application, and
- Ensure the configuration is properly documented

The process is controlled by Nuclear Engineering Procedures (NEP). The scope of the process includes new and replacement items for quality-related applications. The process also describes the relationship between design, qualification, procurement, dedication, and supply. The procedures and process may be applied to items for non-safety-related applications.

Once the need for an item is identified, a determination is made whether an item has previously been identified for use in the specific application. If the answer is no, the design requirements for the item are established. The design requirements may apply to current design and or those required for a design change. Design requirements are identified through review of design documents, equipment walkdown, safety classification data, technical data on form, fit, and function, and design qualification documentation.

Should a replacement other than like-for-like (identical) design be required, the process directs the user to the applicable procedures for continuation of the process. When qualification of design is required for new or replacement items, the process directs the user to the appropriate design qualification methods. Once the design, qualification and description of the item is completed, the process directs the establishment of procurement requirements for obtaining items through the supply process. The Quality Receipt Inspection Process verifies that items specified are those that are procured.

A number of checks and balances exist in the current process. Safety-related material purchase orders are quality records and provide a link to the original equipment design specifications. The material engineering procedures impose the technical and quality requirements that reflect the design of the item to be purchased. The verification that purchase order requirements have been met is accomplished through a combination of receipt inspection, dedication testing and engineering review of test results. The receipt process includes independent quality control overview. ASME code items undergo additional verification by Authorized Nuclear Inspectors (ANI).

The process is audited annually by ComEd Quality Verification to the appropriate requirements of 10 CFR 50 Appendix B. Corrective actions are identified and program revisions are made. (See Appendix II, Process 8)

1.6 10 CFR 50.59 Safety Evaluation Process

The 10 CFR 50.59 review screening and Safety Evaluation process at Byron is controlled by Byron Administrative Procedure (BAP) or Nuclear Engineering Procedure (NEP). The processes are similar in scope and purpose. Differences between the procedures have not been found to result in significant differences or results.

Detailed forms and worksheets are utilized to perform 10 CFR 50.59 screenings and Safety Evaluations. Screenings and evaluations are prepared and reviewed by qualified personnel as defined in the appropriate procedure. Copies of all safety evaluations are sent for off-site review. (See Appendix II, Process 13)

Six sites reviewed the 10 CFR 50.59 process in 1996 and proposed a standardized process.

1.7 UFSAR Update

Byron and Braidwood share a common UFSAR update process because they share a common UFSAR and both sites review and approve all changes. The purpose of the UFSAR Update process is to assure that the UFSAR accurately reflects the current plant status. Anyone can identify a potential UFSAR change and complete an UFSAR change form in accordance with a station procedure. All proposed UFSAR changes are processed through the UFSAR Coordinator.

The UFSAR Coordinator has the responsibility to ensure that the update package is complete, including the 10 CFR 50.59 safety evaluation, marked up UFSAR pages, drawings, and other applicable documents. The coordinator is also responsible for coordinating reviews of the proposed changes and resolving comments. After the UFSAR Coordinator affirms that the update package is complete, the package is forwarded to the appropriate departments for review. All technical change packages are reviewed by the engineering department, including off-site departments to ensure technical accuracy. All change packages are approved by a supervisor of the affected department. UFSAR changes are considered part of the UFSAR after the change package is approved and reviewed by management.

Submittals of UFSAR updates are made to the NRC no later than every 24 months from the previous submittal, and include changes made up to a maximum of 6 months prior to the date of filing. (See Appendix II, Process 19). This represents an approved exemption to the frequency stated in 10 CFR 50.71(e).

1.8 Out-of-Service/Return-to-Service Process

This process provides an overview of the common approach utilized to initiate and remove an equipment Out-of-Service.

Any station personnel may initiate an OOS Request to perform work safely on station equipment or to otherwise maintain and control abnormal configurations. This process is managed through Byron's Electronic Work Control System (EWCS). The main points of the process are:

- Work groups requesting the OOS are responsible to sufficiently define the scope of the work to allow the Operations Department to develop an adequate OOS.
- Qualification requirements are established for individuals who prepare and review OOS. Controlled documents and drawings are used to ensure accuracy of prepared OOS. When controlled drawings are unavailable, the OOS is walked down in the field to ensure accuracy. A second qualified OOS preparer independently verifies the OOS as correct.
- The OOS is reviewed by an SRO licensed operator to identify concerns in the area of Technical Specifications, Containment impact, fire protection/Appendix R impact, and other issues.
- A SRO licensed Unit Supervisor in the Control Room conducts an independent review and weighs the impact of the OOS on the unit.
- A RO licensed Nuclear Station Operator (NSO) reviews and verifies the OOS is correct for the current plant conditions and will brief the Operations personnel positioning equipment and hanging the OOS cards.
- Both licensed and non-licensed operators may place OOS cards. All cards are hung and then independently verified unless waived by the Unit Supervisor per station procedure.
- The Work Group Supervisor is responsible to verify the OOS has been correctly hung and is adequate for the scope of the work.

While in place, OOS are subjected to periodic reviews for potential impact on station operation in accordance with requirements specified in station procedures.

When work is completed, a Return-to-Service (RTS) Request initiates removal of the OOS.

- A qualified OOS preparer reviews controlled documents and drawings to prepare the RTS and determine repositioning requirements for equipment.
- A second OOS Preparer verifies the RTS is correct.
- The RTS is SRO reviewed to identify potential Technical Specifications and administrative requirement issues.
- A NSO reviews the RTS and briefs the operators who will reposition equipment and remove the OOS cards.
- All equipment is repositioned and OOS cards are removed with independent verification unless waived by Unit Supervisor per station procedure.

Independent verification is used throughout the OOS program. There are two OOS preparers and each is responsible to independently review controlled documents and drawings to ensure that the points of isolation and special instructions are correct. Technical Specification, Containment impact, fire protection/Appendix R and other operation impact and issues are independently reviewed by SRO licensed operators. When equipment is positioned and cards are hung during OOS or removed for RTS, two operators are normally assigned to perform independent verification. The review by both the Unit Supervisor and NSO considers potential impacts of the OOS or RTS on the current plant configuration. The Work Group Supervisor is responsible to ensure that the OOS is appropriate for the scope of work to ensure protection of the equipment as well as personnel safety. The periodic review of OOS ensures that the level of plant safety is not degraded by the duration of the OOS. (See Appendix II, Process 20)

1.9 Vendor Equipment Technical Information Program (VETIP)

This process provides a methodology for the control of vendor technical information used for the installation, maintenance, operation, testing, calibration, troubleshooting, and storage of equipment. In compliance with Byron's commitment to NRC Generic Letters 83-28 and 90-03, all vendors supplying critical safety-related components are contacted periodically to ensure that the latest manual revision is in the VETIP system.

The VETIP Coordinator receives all vendor manual information at the station and processes the information. The manual or vendor information is logged and tracked by the EWCS or other database. The coordinator reviews the manual for applicability and determines whether the manual is currently in use at the station.

If the new manual is a revision to an existing manual, the coordinator determines whether the change is administrative or technical and prepares a Vendor Document Comparison Report (VDCR) which summarizes the changes between the different revisions of the manual.

The VDCR and manual are forwarded to the subject matter expert (SME) for review. If the SME finds the changes acceptable, then the SME approves the manual and determines what other station groups should be notified of the manual changes. If station procedures are affected, the manual is forwarded to the affected department's procedure coordinator to initiate incorporation as appropriate.

If the SME or other reviewers determine that the technical changes in the manual are not acceptable, the vendor manual is returned to the VETIP Coordinator for final disposition.

After review and approval by the SME, the VETIP coordinator updates other existing hard copies of the manual and databases. The vendor information and all station review/approval documents are forwarded to Central Files for retention. (See Appendix II, Process 14)

1.10 Document Change Request (DCR)/Modification Close-Out

The Document Change Process at Byron Station is controlled by a NEP and associated site appendix for changes to the station drawings. Plant drawings have been segregated into Critical Control Room Drawings and Non-Critical Control Room Drawings (NCCRD). This split is used to prioritize the drawing revision process with the CCRDs taking first priority. These drawings are prioritized in order to ensure the control room operators have the current drawings showing the plant as-built information.

Training on the work flow for drawing changes is part of the general engineering orientation training. (See Appendix II, Process 7)

1.11 Operability Determination Process

- The operability determination process involves the determination whether certain plant equipment is able to perform its specified function. Operability assessments are performed by engineering when the capability of a Structure, System, or Component to perform its specified function(s) as required by the SAR cannot be unequivocally demonstrated, or where a degraded or nonconforming condition results in a judgment that the equipment is operable but there are remaining concerns or uncertainties.

When an operability issue is identified the Operations Manager, Shift Manager, or Shift Operations Supervisor determines whether the SSC is operable without limitations, not operable, or operable but requires additional evaluation. If the equipment is inoperable the applicable Limiting Condition for Operation Action Requirement (LCOAR) is entered. For SSCs that require further evaluation an operability assessment per station procedures is performed by engineering. Regardless of the results of the operability assessment, the responsibility for control of the operability determination lies with Byron Operations Department. The operability issue can only be closed when it can be shown that the equipment has been restored to meet the original full qualifications or the design bases have been changed. (See Appendix II, Process 18)

1.12 Conclusion

The programs and processes used at Byron to maintain configuration control are developed and implemented in a manner consistent with industry standards and commitments. Identified weaknesses and deficiencies are being addressed by ongoing refinements and, where applicable, expanded training requirements.

Audits, surveillances, and assessments have been conducted by Site Quality Verification (SQV), responsible organizations, and third parties, and these audits, surveillances, and assessments have not identified deficiencies representing a significant breakdown of a quality assurance program. Some programmatic deficiencies in processes and procedures have been identified and corrective action has been implemented to address and prevent recurrence of these deficiencies.

The implementation of these processes discussed in this section provide reasonable assurance that the configuration of Byron Station is consistent with its design bases.

2.0 ACTION (b)

RATIONALE FOR CONCLUDING THAT DESIGN BASES REQUIREMENTS ARE TRANSLATED INTO OPERATING, MAINTENANCE, AND TESTING PROCEDURES.

2.1 Introduction

Byron Station implements a comprehensive procedure preparation and revision process, in accordance with applicable license and Quality Assurance requirements, that provides reasonable assurance that applicable design bases requirements are translated into operating (normal, abnormal, and annunciator response), maintenance, and test procedures. The rationale for concluding that there is reasonable assurance that design bases requirements are translated into operating, testing, and maintenance procedures is based on the following circumstances:

- Original station procedures were developed using the combined construction and operating knowledge of the NSSS vendor, Architect Engineer, and ComEd. In many cases these procedures were tested on actual station hardware prior to and during station startup.

- Subsequent to startup, some procedures have been revised and new procedures have been prepared in accordance with applicable station administrative procedures that implement Quality Assurance requirements. When the plant was licensed, the NRC necessarily concluded that these original procedures provided reasonable assurance that design bases requirements had been translated into operating, maintenance and testing procedures. These Station Administrative Procedures incorporate a number of reviews (checks and balances) which are intended to assure that all applicable design bases requirements are considered prior to the approval for use of each procedure revision or new procedure.

- Operating, maintenance and testing procedures have been implemented in the station for many years. The resulting consistency between expected and actual station responses indicates that design bases information has been translated into these procedures accurately.

- Several procedure reviews and enhancements have been conducted and have resulted in either corroboration of the translation of design bases information or in the enhancement of procedures in this regard. The number and type of procedure deficiencies identified from recent reviews of Corrective Action Records (CARs) did not show significant deficiencies in the translation of design bases information into procedures. For example, review of all station CARs from 1982 through 1996 shows that seven of approximately

1,600 CARs involved inadequacies in the translation of design bases information into operating, maintenance and testing procedures. Those inadequacies were isolated, as demonstrated by their distribution in time and over subject areas.

- Audits and inspections by both ComEd and external agencies have implied, based on their broad, generally applicable findings, that the procedure control and revision processes are structured to provide reasonable assurance that design bases information is accurately translated into operating, maintenance and testing procedures. In the few cases where deficiencies were identified, appropriate corrective actions were implemented.

Each of these circumstances is discussed in more detail below.

2.2 Consistency of Original Station Procedures with Plant Design Bases

Original station testing, maintenance, and operations procedures were prepared prior to startup by the NSSS vendor, Architect Engineer, and ComEd. Operating experience at other stations, vendor equipment requirements, and design bases were all considered in the preparation of these procedures. Many of these procedures were implemented during testing and other pre-startup activities. Formal verification efforts were conducted to ensure the adequacy of the original procedures. This included steps to assure their conformance with the licensing and design bases.

Two significant examples of tests of translation of design bases requirements were (1) a review of pre-operational and startup testing conducted by the ComEd Quality Assurance organization for all safety-related systems, to ensure that all functions described in the FSAR were adequately tested; and (2) a review of operating, maintenance, and testing procedures conducted by the Westinghouse Site Engineering Team and the ComEd engineering staff, to ensure that the procedures fulfilled the Technical Specification Surveillance requirements. The successful conclusion of these and other tests of procedures were relied on by the NRC when it concluded that the station's procedures were adequate when the station was licensed to operate.

2.3 Procedure Preparation and Revision Processes

The procedure preparation and revision processes incorporate several elements that are designed to assure that the applicable design bases requirements are identified and are correctly translated into operating, maintenance and testing procedures. The procedure preparation process, and the personnel qualifications for procedure preparers and reviewers, were established in accordance with original station Technical Specifications. These preparation and qualification procedures remain part of the current procedure process. The station administrative procedures provide additional focus on the use of design bases information sources and appropriate procedure content.

Operations and maintenance procedures are prepared by knowledgeable individuals who are trained where to find design bases information and whom to contact for assistance (Engineering). Procedures are reviewed by qualified personnel in a multi-level/multi-discipline review process. This multifaceted review is a key element of the procedure preparation process.

The procedure preparation and revision processes include the following steps. They provide the checks and balances that help to assure that design bases information is accurately translated into operating, maintenance and testing procedures:

- 10 CFR 50.59 Screening and Safety Evaluation
- On-Site Review (OSR)/Technical Review
- Station Manager Review (for On-Site Review only)
- Verification
- Commitment Preservation

At a minimum, all new procedures and procedure changes are required to undergo a 10 CFR 50.59 screening and technical review prior to approval. Additionally, all new and revised procedures require verification. For procedures which have a possible safety impact, additional reviews are performed by On-Site Review and Station Manager.

An Operating Department improvement initiative is an ongoing process to review and enhance operating procedures for human factors attributes, proper sequencing of steps, and technical correctness. By using field walkdowns as the particular activity is performed, clear presentation is ensured so that verbatim compliance is consistently accomplished by a trained operator. Revisions identified by these reviews are incorporated in accordance with the ComEd Quality Assurance Manual per station procedure.

2.3.1 10 CFR 50.59 Screening and Safety Evaluation

A 10 CFR 50.59 screening is performed on all newly prepared or revised testing, maintenance, and operating procedures to determine whether the provisions of 10 CFR 50.59 apply. If so, a safety evaluation is performed to determine whether the proposed change could involve an Unreviewed Safety Question or a change to the Technical Specifications. The screening checks the procedure change against license requirements and the design bases located therein. Personnel who perform this screening must meet the qualification requirements specified in station procedures for minimum education, training, and power plant experience required to function in this role. These requirements have been found sufficient to assure that procedure preparers and reviewers have the necessary knowledge of design bases information.

2.3.2 On-Site Review (OSR)

New and revised procedures are required to receive as a minimum, a technical review, design bases review, and a department/position review performed in accordance with station procedures. Special procedures, tests, and experiments additionally require an operability review. On-Site

Review is conducted in accordance with procedural guidance that may include considerations of (1) determinations of fulfillment of Technical Specification, UFSAR, and station requirements and commitments ; (2) physical plant safety issues; (3) 10 CFR 50.59 Safety Evaluations; (4) procedural compliance; and (5) radiological concerns. Discipline requirements are assigned to reviewers in accordance with station procedures. Senior OSR participants assign reviewers for the required disciplines.

All procedures which require an On-Site Review are also reviewed for safety impact to the operation of the station. Although the focus of this review is safety, it provides an opportunity for senior managers to identify any concerns regarding the accuracy of translation of design bases information into the procedures.

2.3.3 Technical Review

Newly prepared or revised testing, maintenance, and operating procedures are technically reviewed to confirm technical adequacy and compatibility with existing station design and operation. Technical reviews are performed by personnel knowledgeable in the subject matter and who meet the applicable experience requirements specified in station procedures. More than one technical reviewer may be assigned, however, at least one reviewer is a member of the department for which the procedure is intended. Technical reviews are supported with review guidelines used by technical reviewers. The guidelines direct and include: (1) review of applicable station drawings; (2) determination whether the procedure or revision addresses lessons learned and Station commitments; and (3) impacts on systems, other procedures, other programs (EQ, ISI/IST, etc.), other departments, personnel safety, commitments, safety-related equipment, and station or control room operations. By applying these guidelines, reviewers apply design bases information to the review process.

2.3.4 Station Manager Review

The Station Manager reviews the procedures that require on-site review before approving procedure revisions. This independent review process adds an additional check and balance for maintaining the plant design bases when a design bases review has been completed by the OSR. Irreconcilable disagreements between OSR and Station Manager are resolved by the Site Vice President and Director of Safety Review.

2.3.5 Verification

New procedures require verification in accordance with procedural guidance. Procedure revisions may also require verification as deemed necessary by the appropriate station supervisor. Verification consists of reviewing the prepared or revised procedure per a procedure review check list for applicable attributes to determine if the procedure: (1) conforms to station administrative procedures; (2) meets its stated purpose; (3) includes adequate steps and

information to perform the intended function; (4) assigns equipment numbers identical with labels in the field; (5) specifies required equipment; (6) specifies an appropriate sequence of actions; (7) agrees with expected plant/equipment response.

Human factors reviews are performed on procedures against human factors attributes of the procedure review check list such as; (1) ensuring that the procedure format was as described in the Station Procedures, (2) ensuring that commitments were properly annotated; (3) ensuring that nomenclature was unique and consistent; (4) ensuring that Notes, Warnings, and Cautions preceded their applicable step; (5) ensuring the procedure defined what data was to be entered in blanks, e.g., sign-offs, instrument readings, etc.; (6) ensuring that instructional steps contained only one action; and (7) ensuring that the procedure can be performed in the station as written.

2.3.6 Validation

Many procedures are also subject to validation, that is, some form of simulated or trial use prior to actual implementation. The need for validation by "Table Top," "Walk Through," or the plant simulator is considered during procedure review and is dependent upon the type of procedure or change and safety significance.

The station simulator is used as a tool to validate operating procedures and to provide modification training. All changes to Emergency Operating Procedures (EOPs) require validation using the plant simulator. The simulator is maintained in accordance with station procedures, which delineate maintenance, updates, and testing required to ensure the simulator matches the current station configuration and design. The simulator has been certified for training in accordance with 10 CFR 55.45. The station operators receive training on operating procedures and Technical Specifications, which ensures they operate the station such that all Design and Licensing Bases are satisfied.

2.3.7 Commitment Preservation

Every procedure that involves a commitment identifies that commitment in the reference section of the procedure and specifically identifies the step(s) that implement the commitment. Where an entire procedure satisfies a station commitment, that commitment will be reviewed before the procedure is revised in order to assure that the commitment will not be compromised.

2.4 Experience with Procedures

Procedures have been implemented in the plant for many years and have proven their effectiveness through experience. Some examples of plant evolutions which confirm the adequacy of procedures include routine startup, shutdown, refueling operations, and surveillance testing. In addition, the successful response of the plant to abnormal events and transients, such as reactor trips, provides further assurance of the continued adequacy of plant procedures and their consistency with the design bases. For example, Byron has experienced the following unplanned events which demonstrated that procedures were effectively used to respond to the events.

- In 1995, a feedwater pump tripped with Unit 2 at 100% power. The Unit was rapidly ramped down and the motor driven feedwater pump started per station procedures. The safety-related structures, systems, and components performed per design, significantly reducing the total load rejection required to mitigate the event.

- In 1996, a turbine trip from 100% power resulted in an automatic reactor trip. The resultant volume shrinkage in the steam generators resulted in the Auxiliary Feedwater Pumps auto-starting at their design setpoint which provided the required decay heat removal. All safety-related systems operated as designed.

- In 1996, the generator stator cooling pump inadvertently tripped. The standby pump automatically started, preventing a main generator trip, which would have resulted in a reactor trip. Procedures used were adequate to maintain configuration control of the plant. Affected safety-related operated as designed.

- In 1996, Unit 2 developed a steam generator tube leak. The leak was monitored in accordance with procedures, until the decision was made to take the Unit off-line. Procedures specifically written to shut down a Unit with a tube leak were used adequately.

2.5 Programs Which Verify Procedure Consistency

Several programs have verified the consistency of operating and/or maintenance procedures with other configuration control documentation. Two recent improvement programs are discussed below.

2.5.1 Reviews of Selected UFSAR Sections

In 1996, ComEd performed an assessment of its stations' conformance with their UFSARs. Detailed reviews of selected UFSAR sections were performed as part of this process.

The scope and extent of the reviews were consistent with that of NEI Initiative 96-05, Section 3.1.1. Specifically, the UFSAR sections were reviewed to identify descriptive phrases regarding frequencies for tests, calibrations, etc.; configuration descriptions; descriptions of system operation in different modes (e.g., normal, abnormal, emergency); operating limits; and descriptive functional performance statements. Then, the highlighted UFSAR statements were compared with current plant configuration and operational practices, as implemented in plant specific procedures, administrative control, design analyses, and/or Technical Specifications.

During the period April 15 through August 15, 1996, five systems and/or topics were reviewed for Byron/Braidwood (spent fuel pool cooling, radioactive waste systems, steam generator tube rupture, containment spray, and essential service water). These reviews identified forty-five (45)

discrepancies, which were resolved in accordance with established mechanisms for assessment of operability, reporting, and restoration of conformance. None of the items were safety significant, i.e., met the NUMARC 90-12 criteria:

- Does the difference appear to adversely impact a system or component explicitly listed in the Technical Specifications?
- Does the item appear to compromise the capability of a system or component to perform as described in the UFSAR?
- Does the difference appear to adversely impact applicable licensing commitments?

ComEd formed a UFSAR Process Improvement Team with members from each station and the Corporate office. This team has provided recommendations for UFSAR reviews. It is also pursuing the development of a standard review process for all sites that will ensure that plant changes are appropriately reflected in the UFSAR.

2.5.2 CAR Trends and Data Analysis

Over 1600 Corrective Action Records were reviewed for procedural inadequacy related to design bases information. Seven CARs were identified against procedures related to design bases issues. Some programmatic deficiencies were noted in the area of procedure consistency with the design bases and are being addressed. The CARs were scattered over time and area. The CAR documentation showed that corrective actions have either been taken in a timely manner or are scheduled. In most cases, corrective actions appeared effective because the deficiency did not recur. This experience further corroborates the effectiveness of translation of design bases information into procedures.

2.6 Audit and Inspection Results

Procedure adequacy with respect to the accurate translation of design bases information has been reviewed indirectly through the conduct of audits, inspections, and self-assessments by both ComEd and external agencies. At Byron, procedures and testing processes have not been directly audited to date, but procedure attributes have been evaluated in the audits, inspections, and self-assessments that were performed on safety-related systems and comprehensive department evaluations. With the exception of a few programmatic deficiencies which are discussed in other sections of this response, the results of these activities have confirmed that design information has been translated accurately into operating, maintenance and testing procedures.

NRC Closure Inspections, 94006 and 96003, on the Motor-Operated Valve (MOV) Program found the procedures in place to implement the MOV Program to be adequate.

NRC Inspection 93006, an Electrical Distribution System Functional Inspection (EDSFI) concluded that relevant procedures and tests on the Electrical Distribution System (EDS) were adequate. The EDS surveillances were comprehensive. No violations of NRC requirements were identified during the course of this inspection.

2.7 Conclusion

Based on the formal checks and balances in the procedure preparation and revision process and the required procedure reviewer qualifications; the consistency between expected and actual responses when procedures have been used in routine plant activities and in response to abnormal events; and the results of past audits and inspections, including the resolution of identified problems; there is reasonable assurance that the operating, maintenance, and testing procedures are consistent with the design bases. The results provide the rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures.

3.0 ACTION (c)

RATIONALE FOR CONCLUDING THAT SYSTEM, STRUCTURE, AND COMPONENT CONFIGURATION AND PERFORMANCE ARE CONSISTENT WITH THE DESIGN BASES

3.1 Introduction

The bases for Byron Station's conclusion that the configuration and performance of its structures, systems and components are consistent with their design bases can be summarized as follows: When Byron was licensed ComEd demonstrated to the NRC that Byron's SSCs were configured in accordance with, and conformed with the plant's design bases. Functional performance and configuration integrity were verified by an extensive and comprehensive startup and preoperational testing program. Since then, Byron has modified the physical, and, on a routine controlled basis, the operational configuration of some of its SSCs and conducted maintenance on them. Those changes and maintenance have been conducted in accordance with processes, programs and procedures that were designed to preserve the configuration and performance of SSCs consistent with their design bases. These processes and procedures have been described in Section 1.0 and Section 2.0 of this response.

Corroboration that SSCs are configured and perform consistent with their design bases is provided in several ways. Normal operation of the plant as expected, and responses to abnormal conditions as planned, generate a substantial body of experience that demonstrate conformance of the SSCs with their design bases. A large body of data about SSC configurations also has been developed over the years as various SSCs were reviewed for modification or maintenance, subjected to surveillances and ongoing monitoring related to operation, and inspected by plant personnel, the NRC and third parties. Where SSCs have been found to deviate from their design bases, appropriate corrective action has been taken. These elements are discussed in more detail below.

3.2 Initial Determination That Configuration and Performance of the SSCs Were Consistent With Design Bases

Performance and configuration of SSCs was initially determined to be consistent with design bases as part of required preoperational licensing activities. These activities included preoperational and startup testing, calculations and studies, plant walkdowns, and other verification efforts.

A documented comparison between design documents and SSCs was accomplished during the construction and initial licensing phases of the plant which included a detailed licensing review and hearings. To support the licensing hearing, an extensive re-inspection of key construction items was completed, which provided additional corroboration that the design bases were accurately reflected in the physical plant.

In addition to ComEd activities, the following pre-operational NRC inspections were conducted, which further demonstrated consistency between the design bases and the construction of the station:

NRC Inspection Report 84-51

The inspection that formed the bases for this report was conducted from August 13, 1984 to April 2, 1985. The inspection focused primarily on piping systems installation, NRC Inspection and Enforcement Bulletin (IEB) 79-14, and piping as-built analysis. Four non-conforming items were identified and satisfactorily dispositioned and accepted by NRC during the review prior to issuing the Unit-1 Full Power License.

NRC Inspection Report 84-67

This special report focused on anonymous allegations with regard to questionable design practices by the Architect Engineer (A/E). The results of the investigation did not support the allegations and no violation was reported.

NRC Inspection Report 84-71

The special safety inspection conducted for this report was in response to allegations provided by expert witnesses on behalf of intervenors. Even though some minor programmatic deficiencies were identified, they were addressed and corrective actions were implemented.

NRC Inspection Report 84-88

This special audit was conducted in response to allegations that ANI Inspectors were coerced into signing ASME-related inspection reports. The investigation found some programmatic problems, however, there was no impact on the physical plant. Acceptable corrective action for the programmatic problems was implemented.

NRC Construction Appraisal Team (CAT) Report 85-27

This special audit concentrated on reviewing construction related concerns. Several areas of noncompliance were identified. Acceptable corrective action was implemented and verified by follow up inspections.

NRC Inspection Report 86-27

This special inspection was prompted by allegations from a ComEd subcontractor to the NRC stating that as-built drawings at Byron Station Unit 1 have discrepancies and, therefore, violate the requirements of IEB 79-14. Five instances were found where the difference in values exceeded stated allowable tolerances. The allegations were substantiated, however, the significance of the discrepancies was insignificant. The result of the review was the conclusion that Byron Unit 1 satisfies the objective of IEB 79-14.

As part of the final licensing process, ComEd certified by letters that the Byron Units had been designed, constructed, and preoperationally tested in a way that would assure consistency with the FSAR, the NRC's Safety Evaluation Report (SER), and the Commission's regulations. In summary, the bases for this certification included (1) the extensive design control procedures and practices employed by the NSSS supplier, the A/E, and ComEd; (2) the contributory roles of extensive ComEd quality assurance audits and third party reviews; and (3) extensive inspections and scrutiny by the NRC.

The product of the combined inspection and assessment efforts described above provided assurance for Byron that initial configuration and performance were consistent with the design bases.

3.3 Preservation of the Station Configuration and Performance Consistent with the Design Bases

SSC configuration and performance since initial plant startup have been maintained consistent with their design bases through the implementation of programs, processes, and procedures that control physical and operational changes to the station. Plant configuration and performance can be modified through the design change process, plant maintenance, and operator manipulation of station equipment.

The design change and plant maintenance processes are procedurally controlled as described in Section 1.0 of this response. As was discussed, these processes include numerous reviews, tests, and other checks to ensure the desired result is obtained, i.e., maintenance of station configuration and performance consistent with the design bases. Plant operations are performed in accordance with operating procedures which are maintained consistent with the design bases through adherence to the procedure change process described in Section 2.0 of this response.

3.4 Ongoing Verification of Configuration and Performance of SSCs

SSC performance and configuration are monitored on a routine basis to assure that results consistent with design bases are obtained. Some of the routine performance monitoring activities include plant walkdowns, surveillance testing, post-maintenance testing, post modification testing, and implementation of the Maintenance Rule. Each of these activities is described in more detail below.

3.4.1 Plant Walkdowns

The configuration of SSCs is maintained, in part, by plant personnel during their performance of regular duties. Operating procedures require plant rounds to be performed on a regular basis, during which Operating Department personnel record parameters that indicate whether SSCs are operating or prepared to operate within the design bases. SSCs operating parameters such as pressures, flows, temperatures, vibration, and oil levels are routinely monitored. SSC problems are identified during these walkdowns and are documented on Action Requests (ARs) and/or Problem Identification Forms (PIFs) for equipment deficiencies. Issues potentially impacting equipment operability are brought to the attention of plant management and processed in accordance with plant procedures for assessment.

System Engineers are also expected to perform regular plant walkdowns of their systems to validate system configuration and to identify system deficiencies. Visual walkdowns are done periodically on assigned systems/components. A formal, documented walkdown of each assigned system is accomplished on a quarterly basis including interviews and discussions with shift operating personnel concerning system/component problems. Additionally, formal reviews of system performance (System Readiness Reviews) are prepared on a routine basis to ensure system performance is receiving proper station attention.

The purpose of the walkdowns is to identify problems that could affect system operation and problems already identified that may not have received the proper priority. System Engineers look for unauthorized temporary alterations. They review operator workarounds, temporary alterations on their system for continued applicability, work requests for closure, and modifications for closure. System Engineers use their expertise and knowledge of the system design bases to find potential problems related to configuration control and/or equipment performance.

3.4.2 Surveillance Testing

A comprehensive program of plant testing has been formulated for equipment important to safety. The program consists of performance tests of individual pieces of equipment, integrated tests of the system as a whole, and periodic tests of the activation circuitry to assure reliable performance upon demand throughout the plant lifetime.

Byron Station conducted an extensive preoperational test program based on the guidance of Regulatory Guide 1.68. Reviews were conducted which confirmed (1) that all safety-related system functions described in the FSAR were adequately tested by the preoperational testing; (2) that the initial complement of testing and surveillance procedures fulfilled the Technical Specification requirements; and (3) that important functions tested by the preoperational tests have been carried forward in the ongoing surveillance testing programs.

The Byron surveillance testing programs have been audited for conformance to the Technical Specifications. No significant deviations were found. Byron has begun the process of conversion to the Improved Technical Specifications (ITS). Conformance of surveillance testing procedures to Technical Specification requirements will be reconfirmed as part of that process.

The overall plant testing program includes the Inservice Inspection (ISI) and Inservice Testing (IST) programs required by ASME Section XI and 10 CFR 50.55a. In the Spring of 1994, an NRC inspection report indicated that Byron's IST program scope was weak and a level IV violation resulted. Prior to this inspection, Byron had begun a re-scope of the program in preparation for the 2nd interval code change. A scope review, performed in-house, in conjunction with Braidwood, was completed by the end of June, 1994. Revision 0 of the IST Program basis document, documenting reasons for inclusion/exclusion of components from the IST Program, was completed by the end of 1994. The completion of this scope review adequately implemented the required corrective actions identified by the NRC findings.

Audits, assessments, and inspections of these programs have found them to be in general compliance with all requirements.

3.4.3 Post-Maintenance Testing (PMT) and Modification Testing

The Engineering and Operating Departments are responsible for specifying PMTs required to maintain the plant in accordance with the design and license bases. This process ensures plant performance is maintained in accordance with design bases requirements following plant maintenance work.

The plant modification process requires Modification Design Engineering to identify construction tests, modification tests, and operability test requirements and acceptance criteria for plant modifications. This process is part of the modification process described in Section 1.0 of this response. Construction tests are performed to ensure installation work is performed correctly and in accordance with the codes and standards governing the work prior to integrating with the plant systems. Modification testing ensures the plant change performs as expected when connected into the plant systems, and operability testing is performed to ensure the modified equipment will meet the surveillance requirements in the Technical Specifications. The testing requirements are implemented either in the work package for basic testing, or by special tests prepared by System Engineering for more complex tests. This testing provides added assurance that the modification is installed consistent with the design bases. Audits and inspections have not revealed adverse trends for this process.

3.4.4 10 CFR 50.65 Maintenance Rule Implementation

The maintenance rule is intended to provide reasonable assurance that key SSCs are consistently capable of performing their intended functions. Byron was a pilot plant for implementation of the maintenance rule. From 1994 - 1996 Byron developed and implemented a program for complying with the maintenance rule. Criteria developed for identifying the covered SSCs have been

reviewed and viewed favorably by the NRC. A state of the art database was developed to track the covered SSCs. An NRC pilot inspection was favorable. The maintenance rule implementation and compliance program at Byron uses the guidelines and requirements specified in NRC Reg. Guide 1.160, NUMARC documents 93-01 and 93-02, the ComEd guidelines for the maintenance rule implementation (Rev. 0 dated 1/31/94), and other documents. Deviations from NUMARC 93-01 guidelines are identified in the maintenance rule implementation procedure. Implementation of the maintenance rule provides additional assurance that SSC configuration and performance are consistent with the design bases.

3.5 Operational Experience

Performance of the plant as expected provides additional confirmation that SSCs configuration and performance are consistent with the design bases. Byron's and Braidwood's successful response to equipment failures and transients provides assurance, at Byron, that the configuration and performance of key SSCs have been maintained. Credit is taken for successful response to failures and transients at Braidwood based on the replicate nature of the two plant design bases The following examples illustrate that point:

- In 1988, Braidwood experienced a loss of instrument air to both Units 1 and 2. The reactors were tripped manually due to decreasing steam generator (SG) water levels. All safety-related systems operated as designed.

- In 1994, at Braidwood, a spurious main steam isolation signal caused the Unit 1 reactor to trip due to low-2 SG level. The main steam safety valves, steam generator power operated relief valves (PORVs), and the pressurizer PORVs all operated as designed to mitigate the transient event.

- In 1996, a turbine trip from 100% power resulted in a reactor trip at Byron. Rapid water level shrink in the steam generators resulted in the automatic start of the auxiliary feedwater pumps at their design setpoint to provide decay heat removal. The components of the involved system performed and responded as designed and within expected parameters.

- In 1995, a Byron main feedwater pump tripped with the unit at 100% power. In accordance with station procedures, the unit was rapidly ramped down and the motor driven feedwater pump was started. The components of the involved systems performed and responded as designed. Operator action was prompt and appropriate and contributed in significantly reducing the total load rejection to mitigate the event.

- In 1996, at Byron, the failure of a main power transformer bus duct resulted in a partial loss of off-site power (see LER 454-180-96-0007). Unit 1 was in a refueling outage. Unit 2 was at power. The partial loss of off-site power resulted in a loss of site non-essential service water which degraded the instrument air supply for Unit 2. Unit 2 was manually tripped. All safety-related systems operated as designed.

3.6 Special Verifications and Improvement Initiatives

A number of special verification activities and improvement initiatives have been undertaken for the purposes of (1) examining specific aspects of the plant's conformance with its design bases, and (2) enhancing the ability to maintain conformance on an ongoing basis. These initiatives have included one or more of the following types of activities:

- acquiring original design information and improving its accessibility
- revising or establishing specific calculations which implement the design bases (which facilitates verification on an ongoing basis)
- verifying that plant configuration and performance is consistent with design information
- Establishing monitoring programs to confirm conformance with specific aspects of design on an ongoing basis.

Significant examples are discussed as follows:

3.6.1 Assembling Design and Licensing Information and Improving its Accessibility

The following examples of improved accessibility of design and license information, and many of the supporting calculations which implement the design bases, have further enhanced Byron's ability to maintain plant configuration and performance on an ongoing basis.

3.6.1.1 Design Calculation Turnover, Indexing, and Control

As described in Appendix I to this response, ComEd has implemented a program which transitioned to ComEd the design control function from the NSSS supplier and A/E and developed in-house engineering capability. As part of this effort, ComEd acquired calculations used by the NSSS supplier and the A/E in the design of many of the SSCs important to safety (and in subsequent modifications.) For Byron, 165,000 calculation records have been acquired and indexed to speed accessibility. They are controlled in ComEd's Electronic Work Control System (EWCS) and are available at the station.

3.6.1.2 Reviews of Selected UFSAR Sections

In 1996, ComEd performed an assessment of its Sites' conformance with their UFSARs. Detailed reviews of selected UFSAR sections were performed as part of this process.

The scope and extent of the reviews were consistent with that of NEI Initiative 96-05, Section 3.1.1. Specifically, the UFSAR sections were reviewed to identify descriptive phrases regarding frequencies for tests, calibrations, etc.; configuration descriptions; descriptions of system operation in different modes (e.g., normal, abnormal, emergency); operating limits; and

descriptive functional performance statements. Then, the highlighted UFSAR statements were compared with current plant configuration and operational practices, as implemented in plant specific procedures, administrative controls, design analyses and/or Technical Specifications.

During the period April 15 through August 15, 1996, five systems and/or topics were reviewed for Byron/Braidwood (spent fuel pool cooling, radioactive waste systems, steam generator tube rupture, containment spray, and essential service water). These reviews identified forty-five (45) discrepancies. They were resolved in accordance with established mechanisms for assessment of operability, reporting, and restoration of conformance. None of the items were safety significant; i.e., met the NUMARC 90-12 criteria.

3.6.1.3 Electronic Access to Licensing Documentation

ComEd has made the UFSAR, Technical Specifications and Bases, and the NRC SERs available in electronic form with word search capability. (Byron and Braidwood share an 18 volume UFSAR which was prepared in accordance with Regulatory Guide 1.70, Rev. 2.) This capability is available to all users at Byron and Braidwood on the local-area network.

The improved accessibility of the Design Bases information in the UFSAR, and the supporting calculations which implement the design bases, have further enhanced Byron's ability to maintain plant configuration and performance on an ongoing basis.

3.6.1.4 Improved Technical Specifications (ITS)

The Byron Improved Technical Specifications (ITS) were developed from the NRC issued Standard Technical Specifications for Westinghouse Plants (NUREG-1431, Rev 1, April 1995). The development considered the current licensing bases (CLB) and design bases of Byron Station to modify the NUREG document into a station-specific document. The design and licensing bases of the station licensing SER, UFSAR, design documents, and safety analyses were utilized in developing the station-specific ITS document. Each provision of the Current Technical Specifications (CTS) was reviewed for inclusion into the ITS, in accordance with 10 CFR 50.36(a). Current requirements not meeting 10 CFR 50.36(a) criteria will be relocated to station documents.

As each ITS section and the associated bases were developed, they received a multi-level review. The initial development compared the NUREG to the CTS, the CTS bases, applicable SERs, the UFSAR, and other CLB material. The bases section consists of background for the specification, applicable safety analyses, applicability of the LCO, the LCO, action requirements, and surveillance requirements. After a cross functional review by station departments, an On-Site Review was performed for each individual section.

When all sections were developed and the technical reviews were completed, a multi-discipline review was completed and no safety-significant items were identified. Additionally, a final integrated On-Site Review of the project was conducted. After completion of the integrated On-Site Review, an Off-Site Review was conducted. The Byron Improved Technical Specifications were submitted for NRC review. The NRC review is expected to be completed in 1997.

3.6.2 Revising or Establishing More Specific Calculations Which Implement the Design Bases

3.6.2.1 Setpoint Control Program

The setpoint control program was developed to ensure consistency between design bases and instrument setpoints. ComEd developed a standard instrument database, along with a standard methodology for performing the supporting setpoint calculations. Additionally, controlling procedures were enhanced to ensure translation of required instrument settings into the field and subsequent gathering of performance data for future setpoint adjustments. Significant efforts were undertaken to improve the instrument data base by validating the as-built data, and creating setpoint calculations that accounted for appropriate instrument setting inaccuracies and tolerances within the required design bases. For Byron, calculations were verified or re-performed for Byron setpoints, in order to ensure consistency between actual plant setpoints and channel accuracies with the design bases. Additional calculations were generated for Byron as part of the Improved Technical Specifications (ITS) program.

3.6.2.2 Motor-Operated Valves (MOVs) Program

In order to provide adequate assurance that safety-related motor operated valves would function in accordance with their design bases, the NRC issued Generic Letter (GL) 89-10 and supplements requesting industry evaluation and testing of MOVs. To meet the requirements of GL 89-10, ComEd documented the design bases for safety-related MOVs, reconstituted calculations, established performance requirements, performed comprehensive static and dynamic testing of MOVs against the performance requirements and has adjusted MOV setpoints, modified equipment, and revised operating and maintenance practices as necessary to ensure that safety-related MOVs will reliably perform their intended function under design bases conditions. Ongoing implementation of the program including performance monitoring and trending were established through procedural controls. A program coordinator was established in Engineering to oversee and evaluate MOV test results and ensure ongoing actions are taken as needed to continually validate and assure acceptable valve performance consistent with design bases.

The NRC performed closure inspections (94006 and 96003) which confirmed that the Byron program has meet the objectives discussed above.

3.6.2.3 Electrical Load List Control and Voltage Setpoint Calculations

Prior to the late 1980s, Byron Station used a manual load list to control 480 volt motor loads on the plant auxiliary power system. Although the manual list was adequate for the original plant design, additional loads have been added and system capacities are more closely approached. In 1992, an automated Electrical Load Monitoring System for Alternating Current Loads (ELMS-AC) was put in place to control and track load and system changes and to provide a better tool for analyzing the AC electrical system. Many of the plant loads were walked down to provide more accurate nameplate data or actual test data was used. Design calculations were reconstituted and now serve as the basis from which plant design changes are evaluated. A similar effort was completed for the plant's DC power system in 1992. ELMS-DC computer program is used to track loading on each safety-related station battery. This program, utilizing its databases, calculates cell size per IEEE-485, performs battery voltage profile calculations and may be used in sizing the battery charger. Control and tracking of both databases are governed by ComEd Nuclear Engineering Procedures. This effort provides added assurance the electrical distribution system in the plant is in accordance with its design bases.

New degraded voltage relay setpoint analyses were also determined. These analyses were performed using a load flow program to review all electrical buses down to the 480V level. Tap changes were made to improve the margin between minimum switchyard voltage and the maximum reset of the degraded voltage relays. Supplementary calculations have been performed on motor-operated valves, essential 120VAC, and contactor circuits. These supplementary calculations now serve as part of the station design bases.

3.6.2.4 Fire Protection Program

The Byron/Braidwood Fire Protection (FP) Program and implementation of 10 CFR 50, Appendix R requirements are documented in the Fire Protection Report (FPR). In 1994, a major review of the Safe Shutdown Analysis implementing Appendix R was initiated as part of the station's response to NRC Generic Letter 92-08 "Thermo-Lag 330-1 Fire Barrier" issues. During the course of the review, some analysis errors and discrepancies between analysis and actual plant installations were identified. Some of the discrepancies were determined to be reportable to the NRC under 10 CFR 50.72 and 73. LER 454-180-95-005, was prepared and submitted to the NRC in 1995 documenting the discrepancies reportable per 10 CFR 50.72 and 50.73. As a corrective action, Byron/Braidwood initiated an assessment of the Fire Protection Report to confirm the technical adequacy of the document. An independent assessment of the Byron and Braidwood Safe Shutdown Analyses was performed by corporate ComEd and an independent consultant. This independent assessment recommended further review in selected areas. This independent assessment was evaluated by Byron and Braidwood and led to the development of a project plan to review areas of the FPR. This project plan was initiated in the second quarter of 1996 and is ongoing. It has identified one additional instance of reportable deficiencies which was reported to the NRC per 10 CFR 50.72 and 73. Two supplements to LER 454-180-95-005 for this event were submitted to the NRC in 1996. The LER committed to interim compensatory

actions, which were implemented, and long term corrective actions. Some long term actions require plant modifications to be installed in the future. The ongoing FPR reviews are expected to continue into 1997 until all tasks in the project plan are complete.

3.6.3 Verifying that Plant Configuration and Performance is Consistent with Design Information

Special verifications and resulting programmatic control and/or data improvement initiatives have been implemented over the years as a result of industry lessons learned. Some of the more significant initiatives and the results are presented below.

3.6.3.1 Fuse Control Program and Fuse List

In the 1980s, audits identified weaknesses in the fuse control process at several nuclear stations, including ComEd. As a result, safety-related fuses are being walked down to ensure the installed fuse is consistent with the design. As discrepancies are noted, the installed fuse is evaluated and determined to be acceptable as is or replaced. Similar to instrument setpoints, ComEd developed a standard fuse database (from which the fuse list is generated), along with a standard engineering process for fuse selection and control. For Byron, safety-related fuses are being walked-down in the field for conformance with the design bases, and discrepancies are being resolved, as necessary. This is an ongoing effort with an expected completion in 1998.

- NRC Inspection 93006, an Electrical Distribution System Functional Inspection (EDSFI)

  This inspection concluded that the systems remained capable of performing their design bases functions. The NRC EDSFI reviewed 13 modifications in detail with the conclusion that they evidenced good design control, safety evaluations which were thorough and well-documented, and good post-modification testing. The exception for Byron was the fuse list, as discussed above.

3.6.4 Establishing Special Monitoring Programs to Confirm Conformance with Specific Aspects of Design on an Ongoing Basis

In addition to routine monitoring activities previously discussed, special programs have been implemented and/or diagnostic tools used to provide enhanced monitoring of key system components to provide added assurance performance remains consistent with design bases. Two significant examples include the EDG Reliability Testing and Generic Letter 89-13 programs described below.

3.6.4.1 Emergency Diesel Generator (EDG) Reliability Program

ComEd implemented an EDG Reliability Program in 1993. The EDG Reliability Program requirements are based upon the Station Blackout Rule, Regulatory Guide 1.155, Regulatory Guide 1.9, Revision 3 and NUMARC 87-00. The Program helps ensure conformance with the design and licensing bases for EDGs and the postulated Station Blackout event by maintaining and monitoring EDG reliability over time for assurance that the selected targets are being achieved.

The EDG Reliability Program includes: monitoring EDG reliability against target reliability levels (trigger values), requirements for comprehensive condition monitoring, surveillance testing, maintenance, root cause analysis, problem close-out and information services, and actions required if EDG Reliability falls below the target levels (or exceeds trigger values).

Byron has seen a significant improvement in EDG reliability since the implementation of the program.

3.6.4.2 NRC Generic Letter 89-13

NRC Generic Letter 89-13 required Byron Station to confirm essential service water systems would perform their intended functions in accordance with applicable design bases. System design, testing, and operation records were reviewed. As a result, a program was implemented to monitor heat exchanger effectiveness and overall system performance on an ongoing basis, as required by GL 89-13. Implementation of this program provides added assurance that the essential service waters system can function in accordance with its design bases.

The GL 89-13 program at Byron includes: heat exchanger performance testing, heat exchanger inspections and cleaning, underwater inspection of the River Screen House (RSH) intake structure and Essential Service Water (SX) Cooling Tower Basins, periodic flushing of infrequently used or stagnant lines, water sampling, and chemical addition to control bio-fouling and corrosion.

Recent problems have been identified with the Ultimate Heat Sink (UHS) design bases calculations, inspection acceptance criteria used for the RSH and SX Cooling Tower Basins, and low SX flow to the 1D Reactor Containment Fan Cooler. Ongoing efforts are in progress to resolve these issues. Byron expects that these issues will be resolved in 1997 and Technical Specification change request(s) will be submitted, as necessary.

3.7 Audits and Inspections of Configuration and Performance

NRC Systematic Assessment of Licensee Performance (SALP) summary reports for Byron Station were analyzed with regard to identified strengths and weaknesses regarding processes and procedures that could affect the design bases of the plant. For the purpose of this report, the departments deemed most closely associated with the configuration control process are Operations, Engineering, Maintenance and Safety Assessment/Quality Verification. For a

summary of relevant SALP ratings, see Table 3-1 below. The reports covered the time period from November 1, 1988 through August 17, 1996 and included SALP 9 through 13. Weaknesses identified in these reports that could have affected areas with potential design bases attributes have been addressed by aggressive management improvement initiatives. The effectiveness of these actions are reflected in the improved ratings of engineering and safety assessment/quality verification. The SALP rating summary indicates that the programs and processes in effect at Byron appear to be working in maintaining a high level of professionalism. But, it is also apparent that changes within an organization, even when identified early, develop a momentum of their own and can effect the overall performance of the department. An example of a negative change is in the Operations area where an increased number of personnel errors were observed already in SALP Period 12 and the trend continued into Period 13, resulting in a lowered rating. Conversely a positive trend in engineering was observed in SALP Periods 10 and 11 but did not result in an increase in rating until Period 12 when the positive trend was firmly established in the department.

Table 3-1

BYRON SALP RATING				
	OPS	ENGR	MAINTENANCE	SAFETY ASSESSMENT/ QUALITY VERIFICATION
SALP 9	1	2	1	2
SALP 10	1	2	1	1
SALP 11	1	2	1	1
SALP 12	1	1	1	*
SALP 13	2	1	1	*

* Safety Assessment/Quality Verification was assessed as part of the other functional areas rather than as a separate group beginning with SALP 12.

In 1989, at Byron, Inspection SSI 06-89-01, Byron Station's Safety System Functional Inspection (SSFI) of the Unit 2 Auxiliary Feedwater System, was completed. Some inaccuracies were found in the UFSAR relating to the Auxiliary Feedwater System. Deficiencies were identified in design document retrieval, the UFSAR, and in some design documents. By May, 1990, corrective actions were developed and effectively implemented to prevent recurrence. The conclusion was that the 2B Auxiliary Feed train and associated support systems reviewed were found to be operable and capable of functioning per design requirements. This conclusion was derived from the evaluation of the reviews conducted, assessments made and concerns/deficiencies identified during the inspection. Required engineering and station evaluations were completed.

Byron and Braidwood were constructed under the same procedures and practices, and licensed under the replicate plant option of 10 CFR 50 Appendix N. The three inspection discussions that follow involve inspections completed at Braidwood that apply equally to Byron based on the replicate design. A ComEd Nuclear Quality Programs (NQP) inspection of the Auxiliary Power System was completed at Braidwood in 1992. This inspection focused on operability and readiness of the Auxiliary Power System, including conformance to UFSAR descriptions and surveillance requirements. NQP concluded that the system remained capable of performing its design bases functions.

A ComEd Site Quality Verification (SQV) inspection of the Auxiliary Feedwater System was completed at Braidwood in 1993. This inspection focused on Auxiliary Feedwater System operability and materiel condition, including lubrication, post-maintenance testing, and calibration. SQV concluded that system was generally in good condition and capable of performing all its required functions. Some design bases issues remain open which are being actively evaluated.

A ComEd Site Quality Verification (SQV) inspection of the Control Room Ventilation System was completed in Braidwood in 1994. This inspection focused on all aspects of control room ventilation system readiness, including surveillances, preventive maintenance, and incorporation of vendor information. SQV reviewed all modifications to the Control Room Ventilation system to determine (1) if the modification tests specified were adequate to test the affected system functions, and (2) that those tests were performed as specified and met acceptance criteria. No deviations were found. SQV concluded that the system remained capable of performing its design bases functions.

In 1996, a Quality Verification Surveillance (QVS) was completed which reviewed Configuration Management at Byron. Specifically, SQV evaluated the PIF as it relates to identifying, processing, and resolving configuration management issues. 153 PIFs were broadly categorized into categories for recognition of human errors, procedure or program inadequacies, and design or equipment problems. Safety-related systems were represented in most of the PIFs. An Engineering self-assessment was reviewed and included favorable comments on design configuration controls at Byron Station. The SQV concluded that the majority of individuals contacted were aware of plant configuration as an issue, a large database of data is available with some data reduction challenges, and trending of configuration management issues is difficult. The recommendations from the SQV is currently under evaluation.

3.8 Problem Identification Form (PIF) Trends and Data Analysis

For the purposes of this response a total PIF population of 11,529 was queried, spanning from 1990 to October 1996. The data base was queried for the following key words: "UFSAR," "Licensing Bases," "Design Bases," "Configuration Management," and "50.59." The total number of "hits" identified was 273, or ~3%. Of these 273 design bases PIFs, 37 resulted in LERs.

The percentage of PIFs applicable to configuration control has increased over the last 6 years, as has the total number of PIFs generated, indicating a lower threshold for station personnel to identify problems on a PIF. However, the severity of the PIFs as measured against the generation of LERs associated with these PIFs has steadily decreased from a high of ~33% in 1990 to ~10% in 1996.

3.9 Conclusion

Based on the formal certification which was part of the original licensing process; the preoperational verification and testing activities; the ongoing verification which are provided by plant walkdowns, testing programs, and operational experience; special verifications and programs which have improved access to design bases information and enhanced the ability to maintain conformance on an ongoing basis; and the results of past audits and inspection, including the resolution of identified problems and the implementation of improvement initiatives; there is reasonable assurance that Byron's Structure, Systems and Components configuration and performance are consistent with the design bases.

4.0 ACTION (d)

DESCRIPTION OF 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

4.1 Overview

This section describes the processes used by the station to identify problems, determine the extent and root cause(s) of the problems identified, and report problems to the NRC. It also describes the processes used to resolve these identified root cause(s) through appropriate corrective actions, including actions to prevent recurrence. The Integrated Reporting Program is the primary process used by Byron Station. In addition to the Integrated Reporting Program, this section also addresses the identification and correction of issues through other programs, reviews, audits and inspections.

4.2 Integrated Reporting Program (IRP)

Byron Station established the Integrated Reporting Program (IRP) in August of 1993. Prior to this date, there were other similar processes at Byron that were used to meet the regulatory requirements established by 10 CFR 50 Appendix B Criteria XV and XVI and 10 CFR 50.72/50.73. Using lessons learned, the current IRP has refined and strengthened these processes by combining many of them into an integrated process. The purpose of the IRP is to provide a consistent method for identifying problems and non-conformances, establishing methods of investigating those problems, identifying the root cause(s), developing appropriate corrective actions that will prevent recurrence and providing data that can be used for trending. Several Byron Administrative Procedures describe the Integrated Reporting Program.

When a problem occurs, a person knowledgeable about the problem initiates a Problem Identification Form (PIF). Anyone working for ComEd, including employees and contractors, can generate a PIF. The threshold for creating a PIF is low. Moreover, station management aggressively encourages all site personnel (via continuing training and through employee's supervisors) to document concerns or problems through the IRP/PIF process. Personnel can create a PIF by filling out a paper PIF form. The PIF originators should then notify their supervisor of the existence of a problem and its documentation on the PIF.

The IRP process promptly determines whether reporting or an immediate evaluation of operability are needed. The Shift Manager or SRO Licensed Designee reviews all PIFs regularly to determine whether a nuclear safety or operability concern exists. If such a concern does exist, he takes appropriate action to place the plant in a safe condition. This review is conducted promptly or within the first 24 hours of an event. If additional input from engineering is required to

demonstrate operability, then engineering is contacted. Disposition of an initial Operability Assessment by engineering is recommended within three business days. The Shift Manager or SRO Licensed Designee also reviews for potential reportability of a problem. In addition to NRC regulations, guidance on reportability is provided in the ComEd "Reportability Manual." This controlled manual provides an event driven system of decision trees to aid in reportability determinations and address notifications and reporting.

Each business day, the Regulatory Assurance Department IRP Coordinator reviews all new PIFs with the Event Screening Committee (ESC). Representatives from Operating, Engineering, Quality Control, Maintenance, Site Quality Verification, and Regulatory Assurance are members of the ESC.

The ESC uses the combined knowledge of the group to understand the importance of plant problems. The ESC also reviews PIFs for NRC reportability requirements. The ESC reviews each problem and the adequacy of the immediate corrective actions already taken. The ESC then determines the significance of each event based on the actual or potential consequences and escalates management's attention on significant issues. The ESC classifies PIFs per Station procedures. The ESC determines if the PIF can be closed to actions already performed; or the ESC may determine that further evaluation or investigation is needed. The ESC assigns the PIF to the appropriate department for resolution. Regulatory Assurance Department numbers each PIF and tracks the resolution to completion in a computerized database.

The ESC often closes less significant PIFs to the actions already taken. A PIF closed by the ESC does not require a written response. Those less significant PIFs that are left open by the ESC are sent to the responsible department for resolution. A timely written response to Regulatory Assurance is required for open less significant PIFs. The department head of the assigned department approves these less significant PIF evaluations.

Regulatory Assurance forwards the more significant PIFs to the appropriate department for Root Cause investigation or Operability Assessment. The assigned Root Cause investigator performs the Root Cause investigation. The System Engineering Supervisor reviews more significant PIF root cause investigation reports to determine the appropriate On-Site Review (OSR) disciplines required for the review of the report. The OSR process provides the formal mechanism for reviewing the appropriateness of the investigation results by members selected from a group of qualified individuals listed in the station administrative procedure covering "Certification of Participants to ANSI-Recognized Discipline Standards." Upon completion of the OSR review, the Station Manager performs a final independent review and approves each Root Cause investigation report. The Regulatory Assurance Department assigns tracking numbers to corrective actions that are not complete at the time of the approval of the Root Cause investigation report. If required, a copy of the completed root cause investigation report is sent for Off-Site Review. The report is filed in the Byron Central File.

Regulatory Assurance collects data about each PIF and its resolution in a computerized database. This data is used for PIF trending and/or IRP monitoring.

4.3 Other Processes that Identify Problems

4.3.1 Action Request (AR)/Work Request

Action Requests and Work Requests may be used by anyone in the station to identify hardware problems. ARs are the primary vehicles used to effect repairs and other work on plant equipment. All station personnel may initiate an AR or a PIF on deficient equipment. The AR procedure directs the initiator to notify the Shift Manager or SRO Licensed Designee immediately if the problem could affect equipment operability. The Shift Manager or SRO Licensed Designee then screens the AR for operability to initiate Technical Specification required actions, initiate a PIF, and/or determine if the deficiency is reportable. The Shift Manager or SRO Licensed Designee determines if the deficiency requires either immediate action or action prior to the next meeting of the AR Screening Committee. The Shift Manager or SRO Licensed Designee then approves the AR and forwards it to the AR Screening Committee, a multi-discipline group. The AR is assigned to the appropriate work group. For modifications, an Engineering Request is generated and assigned to engineering for processing under the controls of the modification process. The System Engineering Group reviews ARs for recurring system or component failures and analyzes the data to evaluate system performance.

4.3.2 Engineering Request (ER)

An ER is used as a method of requesting assistance from engineering in conducting problem evaluations. ERs are intended to provide an evaluation of a potential or existing problem. The station procedures provide guidelines on how plant personnel submit technical inquiries, design evaluation and design change requests, and evaluations on Temporary Alterations to the Site Engineering Department. This procedure also provides requirements for processing and resolving of ERs. The ER process assures that design bases issues are properly identified, documented, and prioritized based on their significance. PIFs are generated for design bases nonconformances and processed through IRP.

4.3.3 Document Change Request

Discrepancies between plant documentation and the as-built conditions of the plant can be identified through the as-built DCR process as previously described. An as-built DCR is the mechanism for administratively making a document change based on an existing condition; no field work is performed. Before making the document change, the existing physical plant configuration is evaluated for conformance with the design bases. If the document is correct and the plant is incorrect, an operability assessment is conducted. A modification would be issued to restore the physical plant to its design condition if necessary.

As-built DCRs are reviewed via a 10 CFR 50.59 screening to ensure an unresolved safety question does not exist. The process also employees a check list for reviewing possible changes to the UFSAR, Technical Specifications, and other design bases information.

4.3.4 Nuclear Operations Notifications (NON)

A Nuclear Operations Notification (NON) notifies other ComEd Nuclear sites of a problem or event that has occurred at the Station so that the other sites can review it for applicability. NONs summarize the nature, impact, and significance of the event and are published before the event investigation is completed. The Byron Event Screening Committee (ESC) selects Byron Station PIF(s) that should be published as an NON(s). For NONs written by other ComEd Stations, if the problem appears to be applicable to Byron, they are forwarded to the appropriate department for information and/or applicability review. Should the event be applicable to Byron, a PIF will be generated to track its resolution.

4.3.5 Operating Experience Reviews (OPEX)

The OPEX program is the primary means used to review and evaluate operating experience information for applicability and determine necessary follow-up actions. The Byron Administrative Procedure for this program outlines the method for evaluating operating experience information. This procedure applies to any source of industry operating experience information, such as INPO SOERs and SERs, NRC Information Notices, Bulletins, and Generic Letters. In addition, significant PIFs and Nuclear Operations Notifications from the other ComEd stations are reviewed by the OPEX program for applicability to the station. Operating experience assessment and dissemination is the responsibility of the Regulatory Assurance Supervisor. The OPEX Coordinator is responsible for the specific implementation of the OPEX Program, including assigning evaluations, tracking reports through the system, and ensuring that periodic status updates are provided.

If the OPEX issue is determined to be applicable to Byron Station, it is assigned a priority. A determination is made as to what corrective actions would effectively reduce the risk of occurrence at Byron, and action items are issued to the applicable personnel. ERs and/or PIFs may be initiated for design bases related issues.

In December, 1996, a SQV audit was performed on selected aspects of the OPEX Program. Areas chosen for review were based upon potential impact to personnel safety, nuclear safety, declining performance (where applicable), and industry concerns. The OPEX Program was found to be adequate in preventing consequential events at the station. The audit resulted in one finding on the OPEX program. The issue was found to be deficiencies in procedure adequacy and compliance. Some action items from OPEX reviews did not receive appropriate priority, approval and tracking. However, the issues observed had no detrimental impact to safety-related systems, structures, or components. The corrective actions and audit recommendations are being evaluated for incorporation into the OPEX program.

4.3.6 Operator Workaround (OWA)

An OWA issue is defined as equipment operated in the manual mode when its design is to be automatic; operator action during a transient or normal operations to compensate for a degraded condition that is not a part of the design; or compensatory procedural requirements to perform a task due to degraded equipment. The OWA coordinator assigns a tracking number and prioritizes the OWA. The engineer will develop an action plan for resolution of the issue, and process an AR or ER if required for design changes. ARs are reviewed by the AR Screening Committee and ERs are reviewed by the ER Prioritization Group. Open OWAs are reviewed periodically for aggregate impacts and resolved by the OWA Committee. There are approximately fourteen OWAs with design bases attributes within the scope of the above definition. These OWAs do not negatively impact the design bases of the station.

4.3.7 Technical Alert

The Technical Alert program, initiated in 1994, is a ComEd program for operating experience feedback which identifies lessons learned at one station and makes them available to the other stations. The content of a Tech Alert is intended to provide sufficiently detailed information on emerging engineering issues to be useful for other potentially affected sites. In addition, Tech Alerts address lessons learned, solutions identified, and actions needed to address the issue at other locations.

4.3.8 Quality First Process

The Quality First Program allows Nuclear Operations Division employees and contractors to address concerns directly and indirectly related to quality and safety. Employees and contractors are encouraged and expected to voluntarily raise concerns they may have in the performance of their jobs.

ComEd management has high expectations for the entire Nuclear Operations Division when it comes to quality and safety. ComEd management also expects supervisors and the line management team to create an atmosphere where employees can freely voice concerns. The individual raising the concern may request confidentiality and every effort will be made to assure the confidential status is maintained. Feedback will be provided to the individual raising the concern. If the individual does not agree with the resolution, the issue may be escalated to a higher level.

Supervisors are crucial to the concern reporting process since they are in positions which receive the maximum input from the workforce regarding potential deficiencies and discrepancies. All supervisors are expected to be sensitive to potential concerns, clarify communications, assure mutual understanding, and act upon potential concerns in a timely manner.

4.3.9 Audits and Evaluations

Problems are identified by formal audits and evaluations, necessitating corrective actions where required. Examples of established audits are discussed in the following subsections:

4.3.9.1 Site Quality Verification (SQV) Audits

Since 1990, the SQV organization began incorporating performance-based methodology into its audits. This process change was implemented to enhance and strengthen the compliance based audit approach used prior to 1990. Audits are conducted in accordance with Nuclear Oversight procedures. Procedures and instructions establish the methodology, requirements for planning, staffing, preparing, performing, and reporting SQV audits. Deficiencies found during an audit are documented on a Corrective Action Record (CAR).

4.3.9.2 Surveillances

Surveillances are a process and product utilized by the entire SQV organization. This process is mostly used by the ISEG group. Surveillances typically record results of analysis, including issues relating to programs and processes. Processes which transcend departmental boundaries and emergent issues are often candidates for surveillances. In many cases, surveillances are initiated to assess potential performance issues.

4.3.9.3 Field Monitoring Program (FMR)

Preparation and conduct of field monitoring activities is also an SQV function. Field monitoring coverage is determined by oversight of routine suggested activities, functional department responsibilities and adverse or declining performance indicators. Field monitoring activities are scheduled based upon a graded approach analysis in accordance with procedural guidance. However, the intent of the schedule is to be a flexible tool and changed as deemed necessary. FMRs consist of such activities as daily tours of the control room and witnessing field implementation of operating, testing, or maintenance procedures or sequences. Deficiencies are documented on a Field Monitoring Report and a CAR.

4.3.9.4 Independent Safety Engineering Group (ISEG)

The Independent Safety Engineering Group (ISEG) examines plant operating characteristics, NRC notifications, industry advisories, Licensee Event Reports (LERs), and other operating experience information. Information from plants of similar design is reviewed for purposes of improving plant safety. ISEG personnel also conduct surveillances of unit activities to provide independent verification that activities are performed correctly and human errors are reduced as much as practical and make recommendations for improving unit safety. ISEG performs reviews

in accordance with Nuclear Oversight procedures and SQV instructions. Reviews are documented in surveillances, audits and FMRs. Deficiencies identified during ISEG reviews are documented on a CAR.

4.3.9.5 SQV Trending

The Integrated Analysis Administrator in the SQV organization performs an independent analysis of station performance information from an oversight perspective in accordance with established guidance. Positive and negative trends are reported to station and Nuclear Oversight Department management via periodic reports.

4.3.9.6 Quality Control Program (QC)

The Quality Control program is implemented by Station procedures. Quality Control (QC) activities are focused principally upon Maintenance, with inline work request reviews and field inspection activities receiving the most involvement. QC also performs receipt inspections of quality related materials shipped directly to the site. Discrepant or nonconforming items, such as components, parts, portable test equipment, and inspection and test procedures that are identified in the field are documented on PIFs (see Section 4.2). The QC group trends weaknesses identified during work request reviews and field inspections, and provides written reports to management. PIFs are initiated when adverse trends are identified.

4.4 Other Processes That Determine Extent of Problems

There are several methods used to determine the extent of identified problems and determine the necessary corrective actions which include:

4.4.1 Root Cause Analysis (refer to IRP, Section 4.2)

Root Cause analyses are performed to understand how a significant incident or degradation occurred and provide insight on how to prevent recurrence. Station root cause determination procedures require that the impact of the cause of the event on the other unit/train should be addressed in the safety consequences and corrective action sections of the root cause report.

The Root Cause analysis process starts after the ESC assigns a root cause investigation or Cause Determination Evaluation (CDE), depending on the significance of the issue. Root Cause investigations and reports are conducted per station procedures. Due dates are assigned by the Root Cause team leader per station procedure in order to assure that the timeliness of the root cause investigation is effective. Training is conducted for Root Cause team leaders and root cause experts.

4.4.2 PIF Trending and Data Analysis

Currently, trend analyst(s) review all PIFs and assign trend codes. The trend analyst systematically reviews the data periodically to find adverse trends. When significant adverse trends are identified, a PIF is generated to initiate an investigation.

4.5 Other Processes that Identify and Implement Corrective Action

4.5.1 Corrective Action Records (CARS)

This mechanism is used both by Nuclear Oversight and the SQV organization on-site. A Corrective Action Record is a stand-alone document used to identify concerns or strengths identified via during field monitoring, surveillance and audit activities. The CAR is used for documenting, reporting, follow-up, condition close-out and trending. There are four significance severity levels and three management attention status levels in the CAR program.

4.5.2 Nuclear Tracking System (NTS)

Corrective actions and commitments are tracked via the Nuclear Tracking System (NTS) which allows for dependable tracking, searching, and follow-up. Items tracked within NTS include, but are not limited to, SQV CARs, NRC commitments, Problem Identification Reports, corrective actions, and others.

4.6 Processes Which Determine Lessons Learned

4.6.1 Root Cause Determinations

Station procedures require that root cause analyses be performed for significant conditions. The purpose of the root cause analysis is to identify the fundamental cause(s) of the condition. When the fundamental cause(s) of the condition have been identified, corrective actions can be developed and implemented to prevent recurrence.

4.6.2 Effectiveness Reviews

Corrective actions implemented to prevent recurrence as a result of root cause analyses are reviewed for effectiveness within eighteen months in accordance with station procedures. A station effectiveness review procedure provides guidance if the initial corrective action is found to be ineffective.

4.7 Processes for Reporting Problems to the NRC

4.7.1 Licensee Event Report (LER)

All PIFs are reviewed for reportability and events that meet the threshold of 10 CFR 50.73 are reported as LERs. Guidance is provided in the ComEd Reportability Manual. This controlled manual provides decision trees to aid in reportability determinations and addresses notifications and reporting in accordance with 10 CFR 50.72, 50.73, 50.9, and 10 CFR 21 as well as other regulations. The Summary Tables contained in the Reportability Manual provide a concise encapsulation of the various reportability requirements.

4.7.2 Technical Issues Review Process

The purpose of the Technical Issues Review Process is to review technical issues, particularly those having generic implications, with respect to 10 CFR Part 21, "Reporting of Defects and Noncompliances." The process is implemented through a Technical Issues Review Committee or representative, who coordinates the actions to investigate and resolve technical issues, and provides guidance and/or solutions to engineering and licensing issues, particularly those common to more than one site. Weekly meetings of the Technical Issues Review Committee are held, with all six stations participating via teleconference. Technical issues are identified for review from station events, vendor notifications, design concerns, Nuclear Network entries and other industry and regulatory sources. The participation by each of the six stations and the corporate office provides multiple discipline reviews used in this process. If a reportable issue is identified, the station management and VP-Engineering are involved in the review.

4.8 Process Effectiveness

A review of Byron data regarding the identification and correction of problems (including PIFs, CARs, LERs, etc.) indicates that the processes have been successful in identifying and correcting problems related to design bases conformance. For example, approximately 273 such items were identified from 1990 to 1996 out of a total of 11,529 PIFs; however few items were significant per the criteria of NUMARC 90-12. This performance is believed to be indicative of an adequate sensitivity to issues related to design bases conformance.

Some of the specific process elements described are relatively new (e.g., Technical Alerts), and the roll-up of several predecessor processes into the Integrated Reporting Program (IRP) occurred. However, in general, equivalent processes have been in place throughout the station's history. Audits and assessments of these processes have been conducted by ComEd personnel and by external agencies, including the NRC. In general, these reviews have not identified significant deficiencies in the process procedures, their implementation, or the products. To arrive at these conclusions the following methodology in reviewing SQV audits/activities were used:

To determine the level of field monitoring coverage of design bases/configuration control oversight performed, a sort of potential applicable attributes (ex. Design Control, Engineering Review) for FMRs was performed. 1875 SQV Field Monitoring Reports (FMRs) from 1993 to 1996 were reviewed to identify 146 applicable FMRs. The 146 FMRs that addressed configuration control attributes included both positive and negative observations regarding these attributes. The data indicated that positive observations out numbered negative observations 134 to 59. Thirty five positive observations were due to Byron addressing configuration control concerns identified at other ComEd nuclear plants, and other nuclear plants in general.

A total of 97 SQV surveillances were reviewed spanning from 1993 to 1996. Twenty observations were noted that addressed configuration control aspects. One CAR, which had already been identified in the CAR Data Base, resulted from these observations. A single weakness was identified relating to configuration control. Trending of minor deficiencies identified in the IRP Program, relating to configuration management issues, was considered inadequate.

The SQV Correction Action Record (CAR) Database of 1659 records spanning from 1982 through 1996 was queried based on the key words "Design Bases" or "FSAR". Larger numbers of CARs were identified in earlier years, because lower level "concerns" were included as CARs in those years. The results were then individually reviewed for configuration control aspects. A total of 31 CARs were identified. All CARs had either corrective actions completed or being developed in accordance with the Corrective Action Process. Due to the scattering of the data over a relatively long time period, no significant trend or weakness in any one area could be established.

An additional eleven CARs, five of which had been included in the above query, were identified resulting from a review of 1875 SQV Field Monitoring Reports (FMRs) from 1990 through 1996. As stated above, corrective action followed the identification of a problem and again no specific trend in a particular area could be identified.

There were some concerns about Byron's Temp Alt process reflected in evaluating multiple CARs. In November, 1996, Site Engineering reviewed four concerns documented on several CARs submitted against Temp Alts during 1994-1995. The four concerns were indicative of a lack of recognition of installations that require Temp Alts. The immediate actions included weekly meetings with all the departments involved with the Temp Alt process, to disposition the open temporary alterations. Senior station management places the correct priority on Temp Alts, diagnosis of the problem, ensures proper corrective actions be taken for resolution, and prevents the misuse of the temporary alteration program. Temporary alterations are reducing in number and age, and action plans are in place to address remaining temporary alterations. No further actions are required to prevent recurrence of these concerns.

In 1995 a Joint Utility Management Audit (JUMA) was performed to evaluate organizational effectiveness of the Byron Site Quality Verification organization and to evaluate effectiveness of the oversight of the Engineering/Design activities at Byron Station. The audit results indicated

that the SQV organization is effectively meeting site management expectations for problem identification, serving as station conscience and pursuing continual improvement in station standards of performance.

In December, 1996, an SQV Assessment was performed on Byron Station's Site Quality Verification organization. The assessment evaluated the effectiveness of the SQV organization in meeting their roles and responsibilities and performing several functions, (1) audits, (2) ISEG, and (3) Integrated Analysis/Trending. In addition the functional areas of Quality Control and On-Site/Off-Site Review were assessed. The assessment focused on the organization's ability to identify, communicate, resolve, and when warranted escalate issues to appropriate levels of management for resolution. The assessment team used a performance based review approach. The results of this assessment indicate that the Byron Quality organization is effectively identifying and communicating issues and maintaining independence. Performance of SQV has been consistently good over the evaluated period of the last two years. Corrective actions and recommendations are being evaluated for implementation.

A review of the station's data regarding identification and correction of problems (including PIFs, CARs, LERs, etc.) indicates that the processes have been successful in identifying and correcting problems related to design bases conformance.

4.9 Conclusion

Based upon the above discussion, there is reasonable assurance that processes have been generally effective in the identification of a wide-spectrum of problems, including those related to design basis issues. Reported problems are screened for significance. Those problems identified as significant are investigated to identify the fundamental causes of the problem. Actions to determine the extent of problems are required by procedures. Corrective actions, based upon the fundamental causes of the problem are implemented to prevent recurrence. The SQV function has been independently assessed and found to be effective, broadbased, independent and intrusive. Problem reporting to the NRC has been appropriate and acceptable. There is reasonable assurance that the processes for identification of problems and implementation of corrective actions are capable of identifying, correcting, and preventing the recurrence of significant problems with the design bases.

5.0 ACTION (e)

EFFECTIVENESS OF CONTROL PROCESS AND PROGRAMS AT BYRON NUCLEAR GENERATING STATION

5.1 Introduction

Reasonable assurance of the overall effectiveness of Byron Station's processes and programs for maintaining consistency between the plant's configuration and its design bases is demonstrated by integrating the information already presented regarding plant programs, the rationale for the effectiveness of those programs, and the ongoing improvement of those programs through operation of the corrective action program. As detailed in the response to Action (a), above, the station has a complete set of processes and programs that are designed and implemented consistent with the ComEd Quality Assurance Manual and industry standards. These processes and programs provide reasonable assurance of consistency between the plant's configuration and its design bases. Moreover, as described above in the responses to Actions (b) and (c), the station has reasonable assurance that its operating, maintenance and testing procedures accurately reflect the plant's design bases and that the plant's structures, systems and components are consistent with their design bases. Finally, as discussed in the response to Action (d), the station implements an Integrated Reporting Program which provides a consistent method for identifying problems and non-conformances, establishes methods of investigating significant problems, identifies the root cause(s), develops appropriate corrective actions that will prevent recurrence and provides data that can be used for trending. Taken together, this information provides reasonable assurance that the station's processes and programs are effective overall in maintaining the configuration of the plant consistent with its design bases.

This conclusion is corroborated by considering a cross-section of that information which shows that the following five elements of an effective program are satisfied: (1) consistency with design bases at the time of licensing; (2) controls in programs and processes that have been implemented since licensing to assure that consistency with the design bases is maintained; (3) improvements to the availability and adequacy of documentation and improvements to programs and processes to control changes to them; (4) verification of consistency between plant configuration and design bases through self-assessments, NRC inspections and third-party reviews; and (5) continuation of activities that assure ongoing consistency between the plant and its design bases. Each of these five elements is discussed in detail below.

Further reinforcement for these conclusions is provided by management's ongoing communication of its expectations, its policy of providing continuing training for its workers, its sensitizing of the work force to the importance of knowing and understanding the plant's design bases, and its insistence on holding individuals accountable for preserving consistency between the plant and its design bases. Generally applicable management initiatives encourage the identification and correction of off-normal conditions, including deviations from design bases. These initiatives also provide appropriate employees with the tools to know the station's design bases and to

understand their role in ongoing plant activities, and they establish an appropriate level of awareness of the importance of maintaining and operating the plant consistent with its design bases. It is this expression and implementation of management's standards that supports the factual conclusions reached in this evaluation.

Finally, the station concludes that the comprehensiveness of the process of developing this response to the NRC's 10 CFR 50.54(f) request, which included an integrated overview of the site's Configuration Management processes, provides additional support for the conclusion that the station is consistent with its design bases.

5.2 Consistency with the Design Bases at the Time of Licensing

When Byron Station was licensed, the NRC necessarily found that it was consistent with its design bases and that the station's processes and procedures should enable the plant to be operated and maintained consistent with its design bases. The quality and availability of design bases information available to the station at that time was consistent with the contemporaneous regulatory requirements. Since Byron was licensed after TMI, the requirements of NUREG 0737 were incorporated into the original licensed design.

As described in the response to Action (c), ComEd certified that Byron has been designed, constructed, and preoperationally tested in a way that would assure consistency with the Final Safety Analysis Report (FSAR), the NRC Safety Evaluation Report, and the commission's regulations; and that the Technical Specifications accurately reflected the as-built plant and the FSAR. These certifications were based, in part, on extensive ComEd quality assurance audits, NRC inspections, and third party reviews; and included detailed verifications of the initial complement of plant operating, maintenance, and testing procedures for consistency with the licensing documents and the physical plant.

5.3 Control Implemented Since Licensing to Assure Ongoing Consistency with the Design Bases

Configuration control processes used at Byron Station have been described in the response to Action (a). The rationale for their effectiveness has been explained in the response to Actions (b) and (c). These processes and programs have evolved in effectiveness over time as enhancements were identified and implemented through application of the Integrated Reporting Program (See Section 4).

The effectiveness of these processes and programs in maintaining the plant consistent with its design bases stems from both their structure and their implementation. With regard to their structure, this response explains how the procedures that implement the processes and programs are formalized and incorporate cross-functional reviews which provide checks and balances. Implementation of the procedures is subject to the station's generally applicable expectations for self-assessment, to management oversight, and to requirements that all individuals involved have the appropriate training and experience.

Moreover, the effectiveness of these programs and processes has been demonstrated repeatedly by ComEd audits, reviews and assessments, NRC inspections and SALP reports, and third party assessments. Where deficiencies have been identified they have been responded to effectively.

In summary, based on the proven processes summarized previously, and the strong procedures program as outlined in Section 2.0, Byron concludes that there is reasonable assurance that maintenance and other design change activities have been and are consistent with the design bases and that the plant has maintained configuration control during operation.

5.4 Verifications As Part of Normal Plant Activities

As described in the response to Action (c), Byron is subject to detailed walkdowns by operations personnel, and system engineers, and to a comprehensive surveillance testing program which meets the Technical Specifications and includes the Inservice Inspection (ISI) and Inservice Testing programs required by ASME Section XI and 10 CFR 50.55a. As described in the Appendix II and the response to Action (c), design and configuration control processes require rigorous post-maintenance and modification testing. Finally, as described in the responses to Actions (b) and (c), performance of the plant as expected in response to normal plant evolutions, and to equipment failures and transients, provides additional confirmation that plant procedures and SSCs configurations and performance have been maintained consistent with the design bases.

5.5 Enhancements to Documentation Availability and Adequacy and to Configuration Control Programs and Processes

Improvement initiatives have already been discussed for enhancing the availability and adequacy of design basis documentation and to programs and processes to control plant consistency with its design bases. Of particular note are the acquisition of original design bases information from the architect engineer to improve the availability of that information to the site, training initiatives for operations, maintenance and engineering, and reviews of calculations by the Chief Engineer Organization. Design bases information has been indexed to enhance its retrievability.

5.6 Verification of Design Bases Conformance by Audits, Assessments, and Inspections

Ever since the station was licensed to operate, it has been subject to verifications of the consistency of the plant's configuration with its design bases. During the normal course of operations, operations, engineering, and maintenance personnel walk down areas of the plant, conduct a comprehensive surveillance testing program to meet Technical Specifications and the requirements for Inservice Inspections and Inservice Testing. In addition, the station undergoes self-assessments, QA assessments, NRC inspections, and third party reviews that have addressed the consistency between the plant's configuration and its design bases as discussed below.

5.6.1 Engineering

Engineers and designers perform at least two self-assessments per year. The self-assessment is an integral part of Byron's overall quality program. A representative sample of those self-assessments is provided by the two examples discussed below:

5.6.1.1 Site Engineering Self-Assessment Final Report September, 1994.

This report was chosen since the group had been substantially reorganized and assumed new duties and responsibilities after the decentralization of engineering functions. In order to sustain the continued excellent performance of Byron Station, an assessment team from the Site engineering staff and other interfacing organizations was formed to assess the performance of Site Engineering. The assessment team consisted of four senior engineers from the engineering organization with a member from SQV providing counseling and validation of the assessment process. The assessment team reported to a steering team of senior Byron personnel. Approximately forty personnel from various organizations were interviewed for this assessment. The assessment was performed in June, July and August of 1994. The following topics among others were assessed:

- Long term degraded equipment and resultant workarounds
- Adequacy of operability assessments
- Adequacy of root cause analyses
- Process for evaluating design issues

The self-assessment team found that the Site Engineering Organization at Byron was focused on supporting the excellent performance of Byron Station by controlling the design activities being performed by outside organizations and performing a larger percentage of engineering work in house. The groups were progressing in the development of the skills, tools, processes, and facilities to perform a large percentage of the engineering work in house. The design responsible organization concept provides a mechanism for Site Engineering to control the design activities being performed by outside organizations and to control design calculations on site.

Some of the weaknesses identified were a lack of professional depth, particularly with respect to technical reviews, limited access to vendor manuals, system/component descriptions and calculations. The assessment found that sometimes the engineers did not know what design inputs were available.

The findings of the assessment resulted in aggressive improvement initiatives particularly in the area of training and building professional depth in the engineering organization. The effectiveness of the improvements is demonstrated by the NRC SALP Summary Reports which have rated the Byron Engineering Organization "1" for the last two SALP periods, as discussed in Section 3.7.

5.6.1.2 Modification Design Self-Assessment, First Semi-Annual Assessment Meeting Report 1996

This summary report was chosen to reflect the performance of the matured organization. The report reviewed, assessed and documented 95 individual self-assessments. Personnel were encouraged to identify subjects within their area of responsibility and perform a self evaluation. The following conclusions and, where appropriate, corrective actions were reached as a result of these assessments:

Despite the improvement initiatives implemented from the 1994 Self-Assessment, Site Quality Verification and the NRC had criticized the quality of the engineering calculations. The results of this critique were noted in the SQV exit meeting and an NRC Notice of Violation.

Site Engineering has taken the following actions:

- Training was conducted for all engineers performing calculations.
- The Chief Engineers from the corporate office conducted a design review of the calculations listed in the Notice of Violation (identified in SALP 13) and others. This review identified several areas where improvements in the performance of calculations were warranted.
- The original design calculations for various disciplines are on-site and have been indexed. This indexing provides greater accessibility for the engineer.
- The Integrated Reporting Program is used to track calculation concerns.

5.6.2 Site Quality Verification (SQV) Audits and Surveillances

The internal auditing process at Byron has been in place throughout the design, construction ,and operating phases of the plant. The objective of the SQV Audit Process is to cover all aspects of the ComEd QA Program as well as 10 CFR 50, Appendix B.

Section 4.8 describes the methods that were used in concluding that Byron has an effective quality verification program and that this program has identified individual items of non-conformance regarding the maintaining of configuration control. No significant trends or weaknesses in any given area could be established from the reviewed data. The individual items of non-conformance were isolated, as demonstrated by their distribution in time and over subject areas. Additionally, corrective actions were implemented following the discovery of individual non-conformances.

5.6.3 Nuclear Regulatory Commission (NRC)

Commensurate with their mission to assure the public of safe operating conditions at nuclear plants, the Nuclear Regulatory Commission (NRC) conducts various reviews and inspections. NRC Systematic Assessment of Licensee Performance (SALP) summary reports were analyzed with regard to findings that could be associated with the design bases. The reports covered the period from November 1, 1988 through August 17, 1996 and included SALP 9 through 13. During this period the NRC did not identify adverse trends affecting design configuration or design bases. This conclusion is substantiated by the consistent excellent rating that is reflected in the SALP summaries.

To summarize, the oversight programs at Byron Station as discussed in Sections 5.6.1 through 5.6.3 are diverse and cover a wide range of aspects associated with maintaining the Design Bases. Based on the number of audits and the number of years covered, essentially every aspect of the configuration management process has been examined. Where problems have been identified, corrective action has been implemented. The conclusion is that the process to identify configuration control problems is effective at Byron.

5.7 Continuation of Design Conformance Activities

Byron Station's processes and procedures for assuring consistency between the configuration of the plant and its design bases are ingrained in the station's work ethic. Management's ongoing communication of expectations and the ongoing implementation of worker training provide additional assurance that the effective implementation of these programs and processes will continue. Similarly, continued application of the station's corrective action program provides assurance that these programs and processes will also continue to be enhanced. Feedback on the effectiveness of these actions will continue to be provided by self-assessments, QA assessments, NRC inspections and third party reviews of the station's activities that are designed to maintain the plant's configuration consistent with the design bases.

5.8 Conclusion

As described above, Byron's processes for assuring that its plants are configured and operated consistently with their design bases have evolved over time and are still evolving. Each important component of the configuration control process has been reviewed and evaluated several times through the conduct of internal audits, NRC inspections and third-party activities. Where deficiencies were found, the extent of their impacts was determined and comprehensive corrective actions were taken to eliminate problems and to prevent their recurrence. Over the course of time, the effectiveness of these corrective actions was demonstrated by the declining number of significant discrepancies. Current programs and processes have been determined to be adequate to maintain each plant consistent with its design bases. Based on the foregoing discussion of the effectiveness of control process and programs, there is reasonable assurance that Byron Station's programs and processes have been effective overall in maintaining the plant consistent with its design bases.

Appendix I ComEd Organizational Restructuring to Improve Byron Station's Ownership and Control of the Design Bases

1.0 Role of ComEd Engineering in Design Bases Management

The Station Engineering Organization plays a significant role in controlling, maintaining, and ensuring conformance with design bases. The role Engineering has had in support of station activities has transitioned over time as stations moved from construction to operation. Self-Assessments conducted in the early 1990s pointed to a need to further transition the role of Engineering to one with a more active focus directly at the station. Transition of major responsibilities to Engineering and the role of Corporate versus Site Engineering in assuring design bases conformance is described below.

1.1 Transition of Design Control and Engineering In-House Development

ComEd's historical approach to design had been a Corporate Engineering Department producing design changes, and analysis by predominately managing architect engineering (AE) contracts from the General Office (essentially a Category 3 organization as described in Section 2.2.3 of NUREG 1397).

Multiple AE's were used by Corporate Engineering to perform work, and the AE Guidebook provided direction to all AE's to assure a common approach. This guidebook formalized the interfaces and communication channels between ComEd and the AE. At Byron, the AE that performed the original design was utilized as the primary AE for design activities after receiving the operating license.

System Engineering and problem solving daily support functions were organized through the on-site Technical Staff which reported to Station Management.

The responsibility for design of the reactor core was centralized at a corporate fuel department. In 1990, a transition was initiated for core design activity to be performed by ComEd personnel, instead of utilizing the NSSS suppliers as in the past.

In later 1992, the Stations were organized under a Site Vice President, with a Site Engineering Department (reporting to the SVP) which established the design authority and accountability on-site.

In late 1993 ComEd conducted a self-assessment utilizing senior individuals from TENERA Corporation. This was done at a time when we had established Site Engineering but had not yet initiated major activities to bring significant work in-house. We continued to rely primarily on AE firms for our design. The AEs also held the majority of the design bases information. Common procedures that had been in place prior to decentralization no longer existed and each site was essentially heading at its own pace for understanding and control of the design bases. This review

identified strategic issues and targeted recommendations to deal with those issues. Key amongst them was the understanding and "owning" of the design. ComEd clearly had to become more knowledgeable in the design, license, and operating bases of the plants. We needed to be in a stronger position to control the design configuration and be proactive in matters that require design information to resolve. The TENERA Report provided recommendations regarding access to and control of design information, and suggested that the first priority should be assigned to efforts required to take ownership of the design and develop in-house capability. It also included a recommendation for development and implementation of a plan for consolidation of design information under ComEd control.

In response to this report, a significant engineering transition began in 1994 to move ComEd into a Category 2 engineering organization (NUREG 1397) by January 1997, and Category 1 by year 2000. An Engineering Vice President position was established. Site Engineering had to be a capable design authority; and it had to hold itself accountable, establish high expectations, and be its own worst critic. The organization that existed at that time lacked many of those attributes because of the high reliance on architect engineers.

A Chief Engineering organization was established in the Corporate Office that was responsible for the establishment of standards, transfer of lessons learned from site to site, oversight of site engineering functions, and the education of the organization as the design authority. The onsite organization was fully integrated into the existing ACAD 91-017 population to ensure that the engineers onsite have a common foundation in engineering fundamentals, plant systems, and site processes.

While we have essentially achieved Category 2 (NUREG 1397) status, our goal is to reach Category 1. We feel our success lies in qualified people, common and controlled processes, and being our own worst critic.

1.2 Relative Roles of Corporate and Site Engineering

As indicated above, the corporate office evolved from being the principle focus for the production of design through architect engineers to an organization that teaches, coaches, mentors, establishes policy, and provides oversight of the design control functions of the site engineering organizations. Dual accountability is established between the sites and Corporate, with Corporate being responsible for technical methods and policy, and the sites being responsible for production and the establishment of priorities. The corporate office does limited production work, primarily in the area of fuel design, Probability Risk Assessment (PRA), and common multi-site projects, e.g., steam generator replacement.

In establishing commonality among the sites in the area of tools and standards, the corporate office procured and implemented the Sargent & Lundy design standards. Common Nuclear Engineering Procedures were established and implemented (and are still in progress); computer codes likewise have been standardized.

One key role of the corporate office is information transfer. Information transfer assists in problem solving, shares information for design modifications of similar components or for similar systems from site to site, and shares the results of assessment and oversight activities. The key information transfer vehicles that have been used are a daily engineering phone call, a Tech Alert program, Chief Engineer oversight of station activities, the Engineering Managers Team meeting, and Engineering Peer Groups.

Tech Alerts - Tech Alerts are prepared and issued by Downers Grove to provide sufficiently detailed information on emerging engineering issues to share lessons-learned, solutions identified, and identify actions needed to address the issue at other locations.

Chief Engineer Oversight Role - The Chief Engineers meet periodically at each site with the Site Engineering Manager and his staff to review detailed design packages. The objective of the reviews is to assure that the design is adequate and to promote the concepts of self-assessment in the engineering organization.

Peer Groups - The Peer Groups provide a mechanism to share lessons-learned, champion consistency on common issues, focus actions on key issues, prioritize activities, and elevate larger issues to the Engineering Management Team. Over 50 groups are active in the areas of management, components, generic programs, general design, and special projects.

1.3 Configuration Management Philosophy

Configuration Management is highly visible at ComEd throughout the Nuclear Stations. The departments at our stations share a responsibility in maintaining Configuration Management. Engineering is accountable for ensuring the design bases is in conformance with the physical plant; Operations is accountable for ensuring the operational configuration is maintained and that operation procedures comply with the design bases; and Maintenance is accountable for ensuring work control processes are conducted in accordance with the design bases.

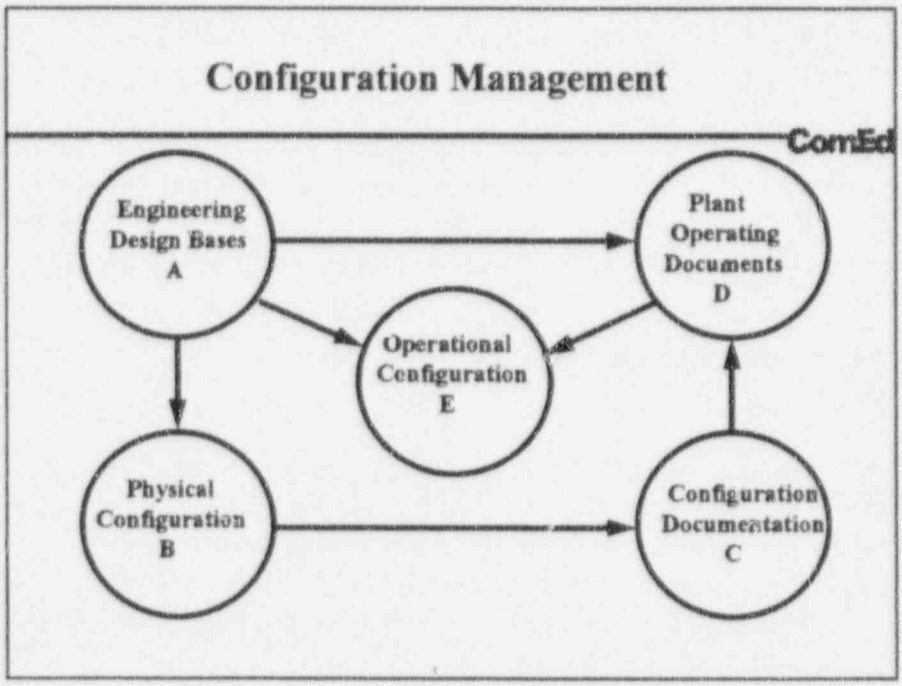
At the corporate level, there is a Chief Engineer, Configuration Management, reporting to the Engineering Vice President. The Chief is accountable for setting policy for configuration management and implementing the policy through a series of common processes and procedures. These common processes are documented in a set of Nuclear Engineering Procedures (NEPs) used across the six nuclear stations.

At each of the six nuclear stations, there is a supervisor in the site Engineering Department who is accountable for configuration control. This supervisor oversees the design change processes discussed in Action (a), and supervises the close-out of the design changes to ensure all controlled documentation (with the exception of procedures) and databases are updated in a timely manner. Procedure update is the responsibility of the owner of the affected procedures.

In structuring our response, ComEd used the following "five ball" model in discussions between station Project Leads and the Project Manager who are responsible for the 50.54(f) letter:

Actions (a), (b), and (c) of the 50.54(f) letter can be directly related to this model. Action (a) is the description of configuration control processes. These are the processes that maintain the design bases in "configuration" with the plant operating documents (A to D link) and the physical configuration (A-B link), as well as with configuration documentation (B-C/and C-D links), i.e., drawings, databases, and reports. Action (b) is conformance of procedures to the design bases (as described in 10 CFR 50.2) (A-D link). And, Action (c) is conformance of the physical configuration and plant performance to the design bases (A-B and A-E links).

Action (d) is also addressed in the above model. When one of the five ball "links" is identified as being in non conformance, ComEd's Corrective Action Programs as described in Action (d) documents the non conformance and initiates corrective action to fix the immediate problem, investigate the cause of the problem, and, if significant, fix the root cause of the problem.



2.0 Self-Assessments Organizations and Departments

ComEd implements programs to provide assurance plant actions are in accordance with design bases. Some of these are required by regulation, such as the quality verification function. Others, such as corporate and site engineering assurance, are self initiated. A description of key organizations and departments and a highlight of their role in providing assurance of design bases conformance is provided below.

2.1 Corporate Engineering Assurance

The Corporate Engineering Assurance Function is part of the Configuration Management organization. The role of this group is to provide technical assurance that the work performed by

Architect Engineers and other contractors is in conformance with ComEd's Nuclear Engineering Procedures and the QA Manual. This is accomplished through periodic audits of the AEs, generally in a teaming arrangement with the Quality Assurance Department.

The Corporate Engineering Assurance Group leads a peer group of the site Engineering Assurance group leaders to provide self-assessment, SSFI, and cross-station evaluations of findings.

Finally, the Corporate Engineering Assurance Group coordinates the generation and reporting of performance metrics for the Engineering Department.

2.2 Site Engineering Assurance

As a result of the NRC Independent Safety Inspection at Dresden in November 1996, which pointed out weaknesses in the oversight of the site engineering activities, onsite Engineering Assurance organizations directly reporting to the Site Engineering Manager are being established. This added assurance function is to provide independent oversight of the expanded accountabilities of the site engineering organization since assuming design authority from the Architect Engineering firms.

The Onsite Engineering Assurance group will oversee the following activities, giving priority to the most risk significant systems:

1. Design Change Activities
2. Operability Evaluations
3. Safety Evaluations
4. Engineering Evaluations
5. Calculations
6. Surveillance Trending
7. Special Test Procedures
8. Performance Improvement Process
9. Licensee Event Reports

The Engineering Assurance Group will focus on the following for the above activities:

1. Verify that the design inputs and assumptions are validated, and if necessary, reconstituted.
2. Verify that the activity is enveloped by the Station's licensing and design bases.
3. Review for operability concerns.

This activity is not a substitute for reviews currently implemented in the existing design control processes. It is intended to be near real-time and concurrent with respect to the engineering activity being evaluated.

2.3 Offsite Review

The Offsite Review and Investigative Function resides at the Corporate Office of the Nuclear Division in the Nuclear Oversight Department's Safety Review Group. Each Site submits documents to Offsite Review in accordance with Section 20 of the Quality Assurance Topical Report (QATR). This includes all operability assessments, Safety Evaluations, and Licensing Event Reports. The Offsite Review for each document requires two participants and an approval signature. As reviews are completed, they are transmitted to the Sites. Reviews may have comments and recommendations or actions assigned based on the completeness of information contained in the document.

In 1996 there were four separate audits of Offsite Review by the Site Quality Verification personnel and one evaluation conducted by the NRC Region III inspectors. In all cases, Offsite Review personnel were determined to be properly qualified and records were maintained for these individuals. Additionally, the audit teams reviewed specific Offsite Reviews with no findings or comments. The NRC inspection had no findings.

The Safety Review Group conducts quarterly self-assessments of its activities. These assessments have helped Offsite Review provide a more in-depth questioning attitude toward Site documents which, in turn, has increased the expectation for greater document quality from the Sites. Offsite Review performs a trend analysis on each Site's submittal and Offsite Review's responses. This information is fed back to the Site management team. The assessment process has also helped Offsite Review understand the need to interface more at the Sites and attend the On-Site Review/Plant Operations Review Committee meetings.

2.4 Role of Corporate Nuclear Oversight and SQV

The Nuclear Oversight Manager manages the Quality Assurance Program, Supplier Evaluation Services and Safety Review. This position reports directly to the Chief Nuclear Officer. He develops, maintains, and interprets the Company's quality assurance and nuclear safety policies, procedures, and implementing directives. He is responsible for the vendor audit program and for ensuring that audits of Corporate support functions are conducted. He is also responsible for conducting a periodic review of the site audit program to assure that oversight of QA Program implementation is effective.

The Site Quality Verification (SQV) Director is responsible for conducting internal audits, surveillances, and assessments of station and corporate activities to ensure compliance with quality assurance and nuclear safety requirements. This position reports to the Site Vice President. SQV monitors the day-to-day station activities involving operations, modification, maintenance, Inservice inspection, refueling and stores through onsite audits, surveillances, field monitoring, and safety reviews.

Appendix II Configuration Control Processes

Background

This appendix summarizes the major, common processes used at all ComEd nuclear stations to control the plant's design bases and configuration, i.e., maintaining the physical plant consistent with the documented plant and with design bases. These processes are designed to ensure the design bases of the plant are maintained or modified as changes are made to the plant as a result of modifications, repairs, or equipment lineup changes. This appendix supports the description of configuration control and design control processes as required for action (a) of the 10 CFR 50.54(f) response. Those processes which are addressed through a corporate procedure (NSWP or NEP) are in place essentially in the same manner at all six nuclear stations.

Flowchart A of Appendix II summarizes the relationships of the engineering design control process and configuration control process as described in Section 1.3.

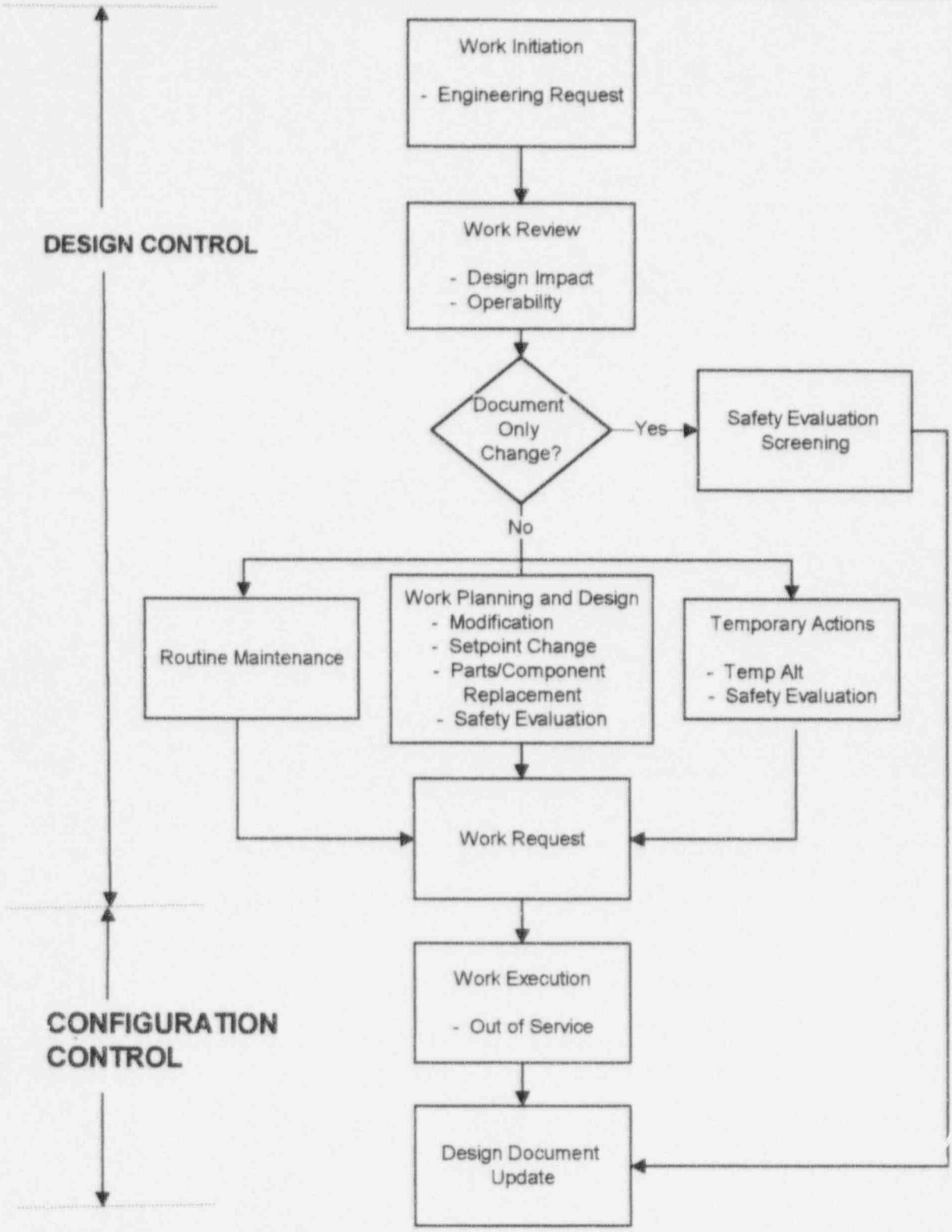
List of Appendix II

Flow Chart A
Matrix

Process Number	Process Description
1	Action Request (AR) Screen Process NSWP-WM-08
2	Roadmap To Design Control Process
3	Design/Document Change Process NEP-04-(Series)
4	Engineering Design Change Process NEP-04-01
5	Modification Work Control Process NSWP G-01
6	Temporary Alterations (Temp Alts) NEP-04-08
7	Document Change Requests (DCRs) NEP-08-03
8	Like-for-Like or Alternate Replacement Evaluation Process NEP-11-(Series)
9	Setpoint Change Request (Station Procedure)
10	Design Basis document (DBD) Update Process NEP-17-01 - DOES NOT APPLY AT BYRON. BYRON HAS NO DBDs.
11	Engineering Software Development and Revision Process NEP-20-01
12	Engineering Change Notices (ECNs) NEP 08-01
13	Safety Evaluation Process (Station Procedure)
14	VETIP Processing NEP-07-04
15	Configuration Control Using EWCS NEP-14-01
16	DBD Development Process NEP-17-01 - DOES NOT APPLY AT BYRON. BYRON HAS NO DBDs.
17	Calculation Process NEP-12-02
18	Operability Determination Process (Station Procedure)
19	UFSAR Update Process (Station Procedure)
20	Out-of-Service/Return To Service Process (Station Procedure)

Flowchart A

Typical Flow of Design and Configuration Control Processes



Matrix of Appendix II Processes

Process Number	Process Description	Procedure Reference	Implements Regulatory Requirement		Configuration Management Model Linkages						
			50.59	50.71(e)	A/A	A/B	B/C	C/D	A/D	A/E	D/E
1	Action Request (AR) Screen Process	NSWP-WM-08				X	X			X	X
2	Roadmap to Design Control Process				X	X	X	X	X		
3	Design/Document Change Process Roadmap	NEP-04-(series)			X	X	X	X	X		
4	Engineering Design Change Process	NEP-04-01	X	X	X	X	X	X	X		
5	Modification Work Control Process	NSWP-G-01	X	X		X	X				
6	Temporary Alterations (Temp Alts)	NEP-04-08	X			X	X	X	X		
7	Document Change Requests (DCRs)	NEP-08-03	X	X		X	X	X			
8	Like-for-Like or Alternate Replacement Evaluation Process	NEP-11-(Series)	X	X		X	X	X			
9 Note 2	Setpoint Change Request	Station Procedure	X	X	X	X	X	X			
10 Note 1	Design Basis document (DBD) Update Process	NEP-17-01				X	X				
11	Engr Software Development and Revision Process	NEP-20-01	X		X						
12	Engineering Change Notices (ECNs)	NEP-08-01				X	X	X			
13 Note 2	Safety Evaluation Process	Station Procedure	X	X	X						
14	VETIP Processing	NEP-07-04				X	X	X			
15	Configuration Control Using EWCS	NEP-14-01				X	X				
16 Note 1	DBD Development Process	NEP-17-01				X	X		X		
17	Calculation Process	NEP-12-02			X	X					
18 Note 2	Operability Determination Process	Station Procedure			X	X			X		
19 Note 2	UFSAR Update Process	Station Procedure	X	X	X	X	X	X	X		
20 Note 2	Out-of-Service/Return to Service Process	Station Procedure								X	X

NOTE 1: This process is not used at Byron or Braidwood.

NOTE 2: Since Processes 9, 13, 18-20 are station unique they are discussed in Action (a), Section 1.0.

Action Request (AR) Screening, Process 1

PURPOSE

Work that needs to be done at ComEd's nuclear stations, is initially identified and documented on an Action Request (AR) except for on-the-spot maintenance, which is initiated using the Electronic Work Control System (EWCS). The AR process is intended to provide all site personnel with a simple and readily accessible process to identify work that needs to be performed. This AR is "screened" to determine the safety classification of the involved equipment, the priority of the work, the work group to whom it will be assigned, and the "type" of work to be performed.

PROCESS DESCRIPTION

The AR screening process begins with a review of a daily Screening Report that captures all of the newly generated ARs. This report summarizes the initial information provided by the initiator of the AR, identifies if the AR is related to a Problem Identification Form (PIF) and is used to determine the appropriate level of controls that are needed to implement the work. ARs can include repairs, maintenance activities, and plant modifications.

A "Screening Committee" determines the appropriate level of controls that need to be applied to the work. The committee brings a required "Knowledge Base" to the table to be used in a consensus determination. This "Knowledge Base" includes:

- Operations - has a current SRO license
- Engineering - is knowledgeable in engineering design and plant design and license bases.
- Maintenance (IM, EM, MM) - is knowledgeable in the division and scope of work among the three maintenance departments.
- Work Analyst - is knowledgeable in work requirements and package preparation.
- Work Control (Scheduling) - is knowledgeable in work scheduling.
- Fix It Now (FIN) - is knowledgeable in FIN Team capabilities.

In addition to the knowledge of the team, the ARs are also screened against the definitions of the work and/or work groups where the work will eventually be performed. The definitions or "types of work" are as follows:

- Modification - A planned change in plant design or operation and accomplished in accordance with requirements and limitations of applicable codes, standards, specifications, licenses, and predetermined safety restrictions. A change to an item made necessary by, or resulting in, a change in design requirements.
- Facilities Maintenance - A minor work activity conducted only on non power plant boundary or equipment. The work will not affect plant or power block structures, systems or components.

- Minor Maintenance - A work activity on Power Plant Boundary Equipment, considered routine and repetitive and within the "skill of the craft" of the maintenance work force. Additionally, minor maintenance requires an initiating work document, does not require detailed instructions, and may be performed without plant scheduling.
- Work Request Maintenance - A work activity requiring detailed instructions and an approval process.

Once the appropriate controls have been determined, the Screening Committee will establish priorities for when the work will be completed. Priority codes and descriptions are as follows:

A **Emergency work** having an immediate and direct impact on the health and safety of the general public or plant personnel, poses a significant industrial hazard, or requires immediate attention to prevent the deterioration of plant condition to a possible unsafe or unstable level. This work must be done immediately.

B1 **Urgent work** that should be scheduled and started within 24 hours.

B2 **Emergent work** that should be scheduled and started within two weeks.

B3 **Emergent work** that should be scheduled and started within five weeks.

C **Routine work** that follows the normal scheduling process.

After the priority has been determined for all work except for modifications, the AR is assigned to the appropriate work group, the documentation is completed by updating EWCS, and the AR is submitted to Work Control/Work Analyst. For modifications, an Engineering Request (ER) is generated and assigned to Engineering for processing under the controls of a modification.

CHECKS AND BALANCES

The first line of defense against potentially performing work with an inappropriate or inadequate level of control is the AR Screening Committee. The knowledge base of the Screening Committee have provided an additional level of confidence to the screening process. By having Engineering participate, it provides a design and licensing bases understanding from people who often reference and interpret the appropriate source documents.

The second line of defense in ensuring that work is performed with the appropriate control is the Work Analyst. Once the initial determination of "type of work" is made by the screening committee, the AR's identified as Work Request Maintenance are sent to a work analyst for further planning and preparation of work instructions. The review and approval of these instructions provides an additional opportunity (the third line of defense) for knowledgeable personnel to evaluate the requested work against the licensing/design bases of the plant and to ensure that no unrecognized design changes are being made.

Additionally, with recent industry and ComEd events that deal with design/licensing basis issues, an increased awareness of the effects changes may have to our plants has occurred. Corporate direction was issued to all sites, directing them to strengthen their evaluation of changes against the definition of a modification and for their potential effect on the design bases of the plant. This

was formalized with the recent issue of NSWP-WM-08, Action Request Screening and represents the third line of defense.

Increased emphasis has also been placed on the definition of Facility Maintenance, Minor Maintenance, and Work Request Maintenance. In each of these types of work, clear boundaries have been provided to maintain the appropriate level of controls. If during the process something requires work to fall outside the predetermined boundaries, the work scope changes or the work scope increases, the work is reevaluated per the initial screening criteria. At that time, the appropriate controls (new or different controls, if applicable) are applied. This fourth line of defense then comes into play because station personnel are encouraged by management and supervision to challenge a work package they believe could be improperly classified.

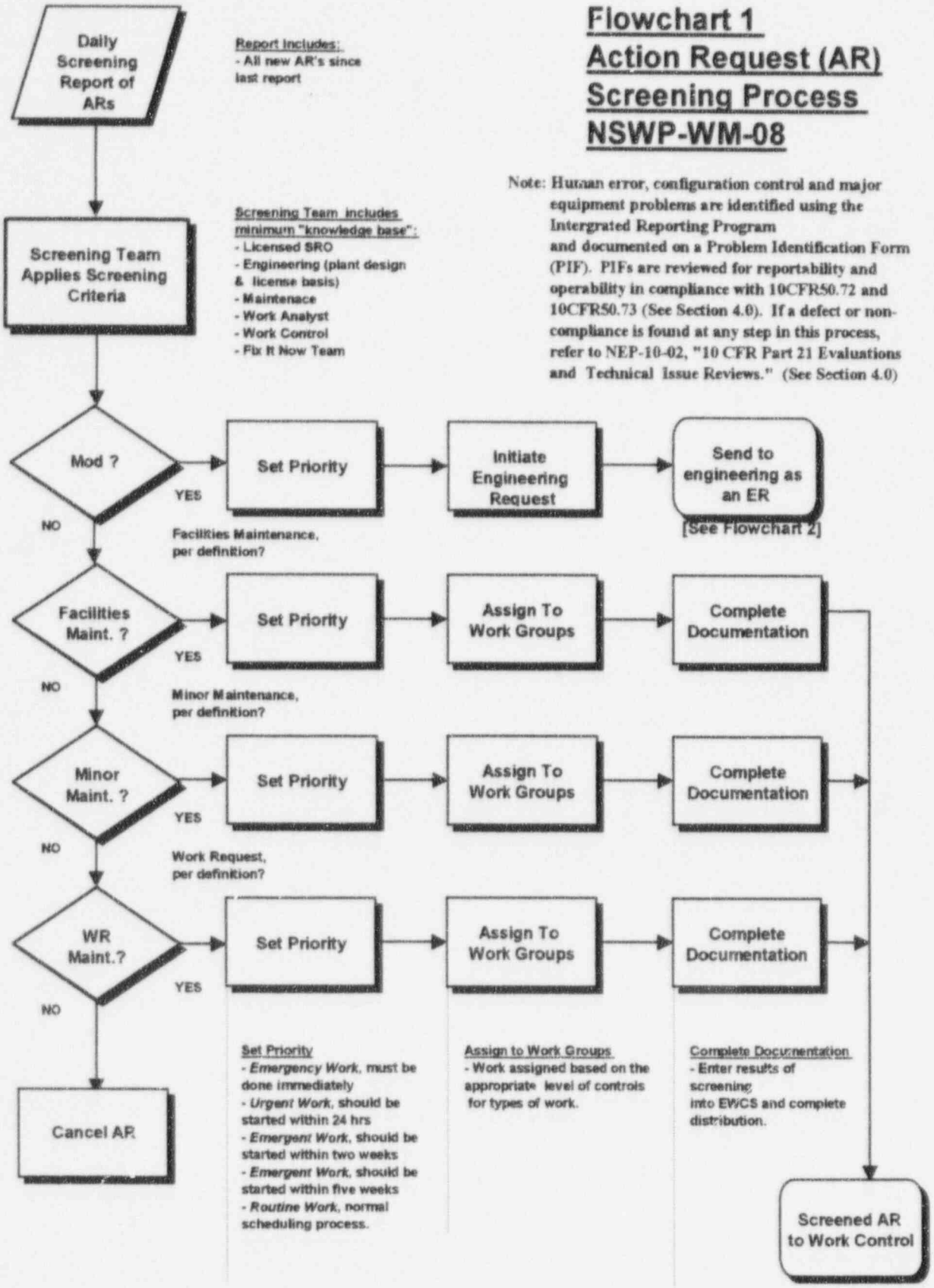
RECENT PLANNED IMPROVEMENTS

Prior to the implementation of EWCS (Electronic Work Control System) in 1994 & 1995, these key screening decisions were made by an Operating Engineer, an experienced Senior Manager with an SRO license. He had a Master Equipment List available to help define safety classification of components and also a set of Piping and Instrument Diagrams color coded to denote code and safety classification. Once safety classification and other decisions were made, including whether the work involved maintenance or modification, the work request was forwarded to the working department for necessary planning, work instruction preparation, inclusion of procedures, etc. This Operating Engineer review was a key control step to ensure identification of work that had the potential to alter the original design. Working department review and approval during the planning phase also provided a secondary control function to ensure that work to be performed did not inadvertently deviate from the plant design.

Since the introduction of EWCS, the methodology has changed somewhat but the intent of the process is unchanged. Decisions on safety classification are now only required on an exception basis as the classification of components has typically been captured in the data base supporting the process. Further, Minor Maintenance teams and Fix It Now (FIN) teams have also been created which have predefined boundaries in which they perform specific types of work. The net result has been a subtle deterioration of the screening function as an effective barrier to inadvertent design changes. In response to this identified weakness, changes have been recently implemented to strengthen the screening process. These changes include the addition of an Engineering participant to the Screening Team and the strengthening of the evaluations performed in accordance with the recently issued Nuclear Station Work Procedure, "Action Request Screening," NSWP-WM-08.

Flowchart 1
Action Request (AR) Screening Process
NSWP-WM-08

Note: Human error, configuration control and major equipment problems are identified using the Intergrated Reporting Program and documented on a Problem Identification Form (PIF). PIFs are reviewed for reportability and operability in compliance with 10CFR50.72 and 10CFR50.73 (See Section 4.0). If a defect or non-compliance is found at any step in this process, refer to NEP-10-02, "10 CFR Part 21 Evaluations and Technical Issue Reviews." (See Section 4.0)



Roadmap to Design Control, Process 2

PURPOSE

This flowchart serves as an overview roadmap of the design control process. It links the major design processes and indicates decision points that determine whether these design processes are required.

PROCESS DESCRIPTION

After the need for a design activity has been identified and an Engineering Request (ER) has been forwarded to Engineering, the first thing that needs to be determined is whether or not the work will be performed internally. If the decision is made to perform the work with an external organization and to delegate design authority to that organization, a Design Interface Agreement (DIA) is required. This DIA establishes procedures among the participating design organizations for the review, approval, release, distribution and revision of documents involving design interfaces. External design organizations are required to meet the requirements of ComEd procedures for modifications in order to maintain design and configuration control.

If the scope of work to be performed involves Nuclear Fuel Services (NFS) this needs to be identified and they need to be brought into the design process. Since the design authority assigned to NFS is retained in the corporate office, and has not been delegated to the stations, their processes, although similar to those described here, are separate, and need to be addressed separately.

If the design involves ASME Section III systems or components, a parallel series of design requirements and processes are required to be performed in addition to the design change process described here. Because these requirements pertain only to ensuring Code compliance, they are not described in more detail.

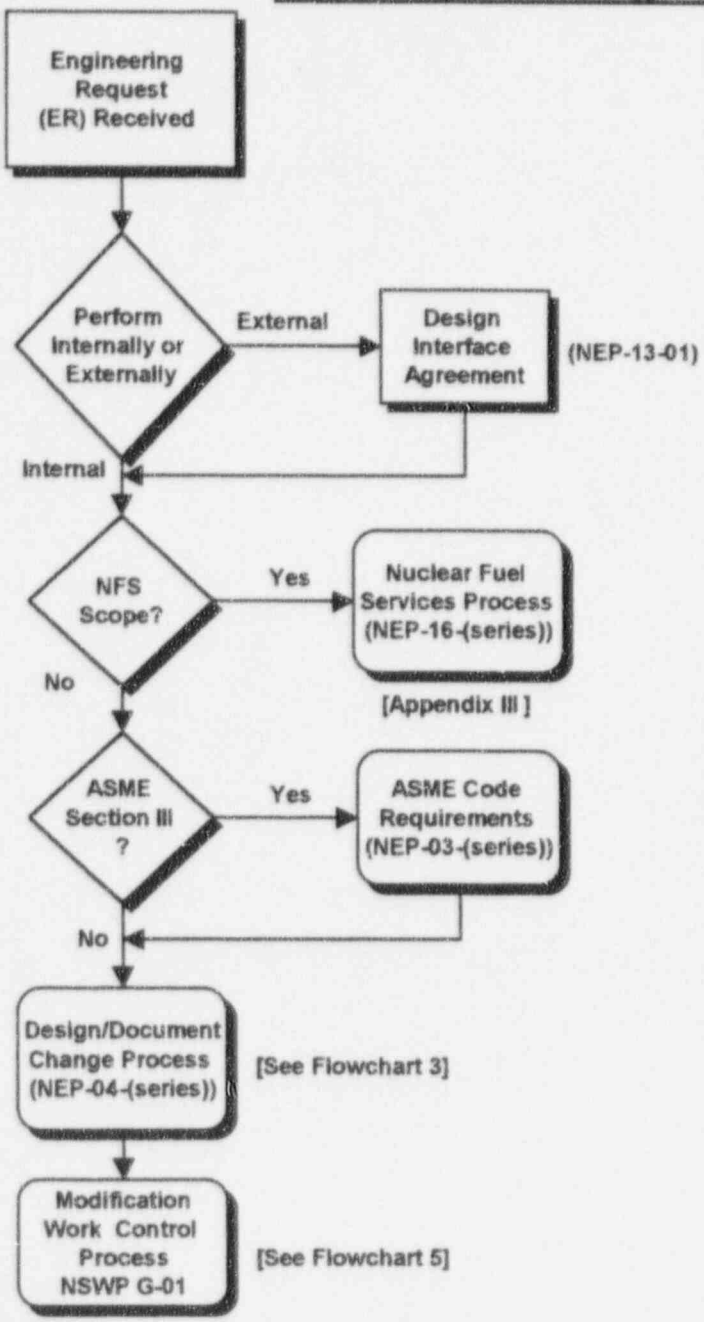
The Design Change Process and the Modification Work Control Process will be described separately in the detailed process descriptions that follow.

Throughout all of these processes and overlaying all of them is the process of identifying and reporting defects and noncompliances. This process applies and can be invoked at any stage and within any of the processes identified here. This process is described separately in more detail.

CHECKS AND BALANCES

The checks and balances applicable to the processes represented here will be described separately in the detailed process descriptions. Human error, configuration control and major equipment problems are identified using the Integrated Reporting Program and documented on a Problem Identification Form (PIF). PIFs are reviewed for reportability and operability in compliance with 10 CFR 50.72 and 10 CFR 50.73. If a design defect or noncompliance is identified, it is evaluated in accordance with NEP-10-02, "10 CFR Part 21 Evaluations and Technical Issue Reviews." The processes are described in greater detail in Section 4.0.

Flowchart 2
Roadmap To Design Control Process



Note: Human error, configuration control and major equipment problems are identified using the Integrated Reporting Program and documented on a Problem Identification Form (PIF). PIFs are reviewed for reportability and operability in compliance with 10CFR50.72 and 10CFR50.73. If a defect or noncompliance is found at any step in this process, refer to NEP-10-02, "10 CFR Part 21 Evaluations and Technical Issue Review."

Design/Document Change, Process 3

PURPOSE

This flowchart serves as a roadmap to the appropriate process to be used in implementing design changes to the plant. At each decision point, a specific process that applies the appropriate level of controls to the change, is chosen. Each decision may be determined through the use of specific definitions, screening questions, and/or lists.

PROCESS DESCRIPTION

Non-Power Block Changes - The first decision point determination is whether the proposed changes can be processed as Non-Power Block Changes. These are permanent changes made to Structures, Systems, and Components (SSCs) that have no impact on nuclear safety, are not subject to NRC regulatory requirements and are not required for the generation of electric power.

Temporary Alterations (Temp Alts) - The second decision point determines if the proposed change is permanent or temporary. Temp Alts are defined as a planned change (non permanent) to the fit, form or function, of controlled operable SSCs, or circuit that does not conform to approved design drawings or other approved design documents. This process is described separately.

Hardware / Documentation Changes - A decision is made to determine the type of permanent change being made. Documentation changes that are clearly administrative in nature, are processed through the As-Built Design Change Requests (DCRs), Setpoint Changes, Computer Software Revisions, UFSAR Revisions or Design Basis document Changes. Each of these processes is described separately.

If hardware changes involve a reload core design, they are processed in accordance with Nuclear Fuel Services (NFS) procedure, "Reload Core Design" (NEP-16-75). This process is described separately.

Other hardware changes and documentation changes that are technical in nature, are reviewed against the definition of equivalent replacements. These include like-for-like replacements or replacements of parts, components, subcomponents, and materials that meet current interface, interchangeability, safety, fit and functional requirements of the original components. This process is described separately.

Changes that are more involved, will be processed as Engineering Design Changes. These include changes to SSCs that are safety-related, subject to NRC regulatory requirements, or are necessary for electric power generation. This process is described separately.

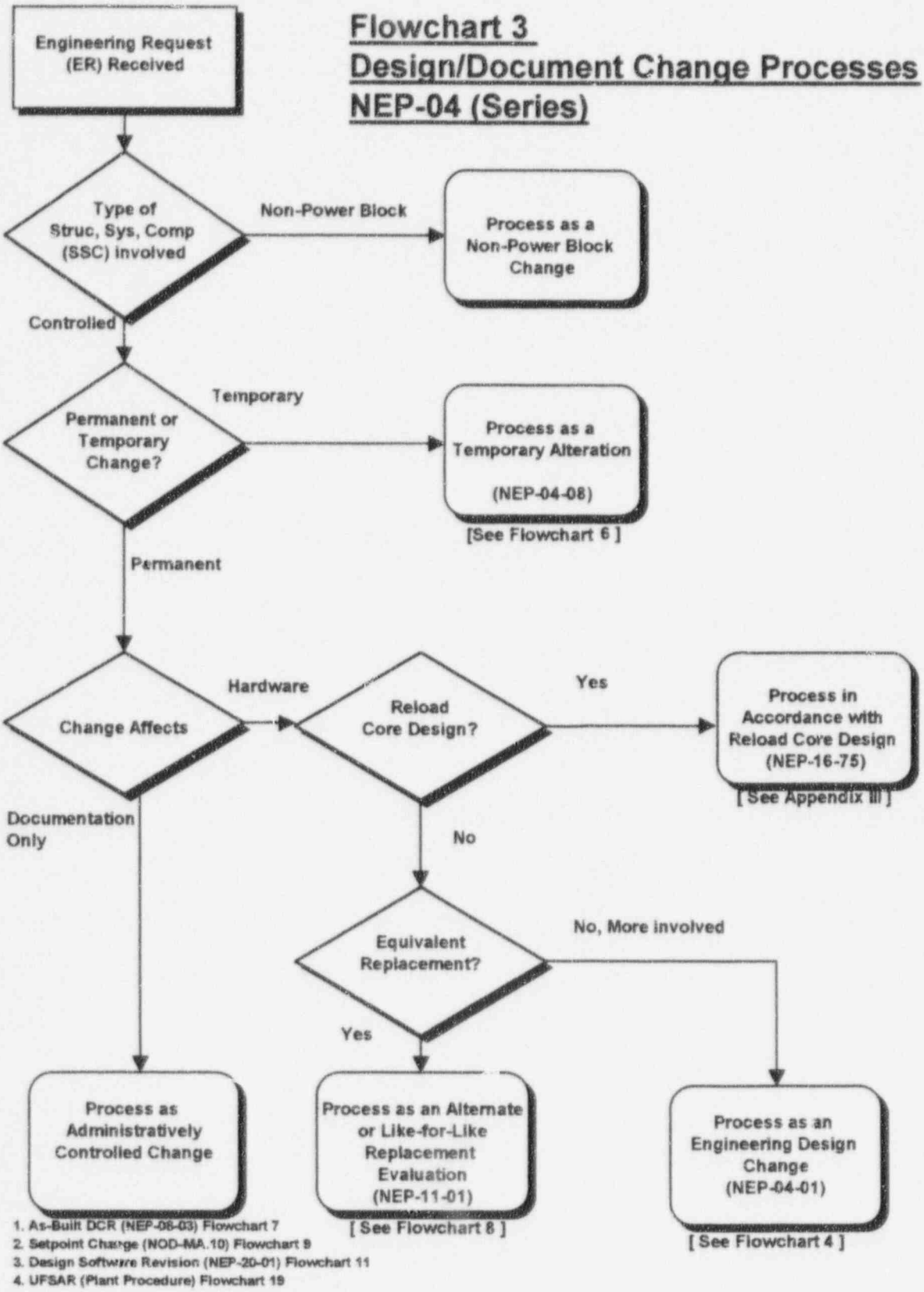
CHECKS AND BALANCES

The checks and balances that apply to the processes represented here will be discussed separately in the individual process descriptions.

RECENT/PLANNED IMPROVEMENTS

In order to reduce the administrative burden of including changes which have no impact on nuclear safety, are not subject to NRC regulatory requirements and are not required for the generation of electric power, ComEd has established a separate process for handling these "Non-Power Block Changes." This process is currently being used at Braidwood, Dresden and LaSalle. This revised process is based on the guidance provided in EPRI TR-103586, "Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants." An Engineering screening review is utilized to determine applicability of this process. Implementation of this process enables ComEd to focus its resources and management on those changes that do have a potential impact on nuclear safety, regulatory compliance or generation of electric power. Improvements in other areas represented on this flowchart will be discussed separately in the individual process descriptions.

Flowchart 3
Design/Document Change Processes
NEP-04 (Series)



1. As-Built DCR (NEP-06-03) Flowchart 7
2. Setpoint Change (NOD-MA.10) Flowchart 9
3. Design Software Revision (NEP-20-01) Flowchart 11
4. UFSAR (Plant Procedure) Flowchart 19

Engineering Design Change, Process 4

PURPOSE

This is the process used to implement "Controlled Design Changes" to the plant. These changes include changes to Structures, Systems, and Components (SSCs) that are safety-related, subject to NRC regulatory requirements, or are necessary for electric power generation. This process provides the requirements for implementing changes that could potentially affect the design bases of the plant.

PROCESS DESCRIPTION

Prior to initiating a planned change to the plant design or operation, ComEd management requires the following prerequisites to be performed before significant resources are expended:

- Approval of technical objectives and proposed conceptual design, including an assessment of compliance with the design and licensing bases,
- Approval of the budget and source of the funding,
- Assignment and approval of the selected design organization, and
- Assignment and approval of the installer(s) and a proposed installation schedule.

After the above prerequisites are met, a Modification Scope Meeting is held. This meeting brings together appropriate Engineering, Operations, Maintenance and Support personnel to review the scope and schedule for the modification, define responsibilities, determine deliverables, review the preliminary design, identify and confirm design inputs, perform a pre-design walkdown and resolve or identify potential concerns or problems. Once the ER is approved as a Controlled Design Change, a Design Change Package is created through Electronic Work Control System (EWCS). A Work Request (WR) is initiated that will be used to implement the required work.

The design is then processed through a series of individual steps that include a scoping activity, field walkdowns, preparing Design Input Requirements (DIRs), engineering calculations, documents, and 10 CFR 50.59 safety evaluations. The DIR defines the major technical objectives, constraints and regulatory requirements that govern the development of the design. It addresses design input categories and serves as a common reference point for the preparation of the more detailed design related documents such as drawings, specifications, calculations, analysis and test specifications. Once the Design Change Package is completed, a final Technical and OnSite Review is initiated that provides for interdepartmental reviews.

After the reviews have been completed, the Design Change Package is issued for Work Instruction preparation as the first step in the Modification Work Control Process. This process is described separately.

In all cases, the design and engineering activities described in these processes are implemented at ComEd by individuals who have been trained and are qualified to perform these functions. These individuals are trained and their qualifications are documented in accordance with the NEP-15-XX series of procedures. These procedures address and comply with the requirements of ACAD 91-017, "Guidelines for Training and Qualification of Engineering Support Personnel," Rev. 1 and ANSI/ANS 3.1, "Selection, Qualification and Training of Personnel for Nuclear Power Plants." This topic is addressed in more detail in the special section of this response that addresses training and qualification.

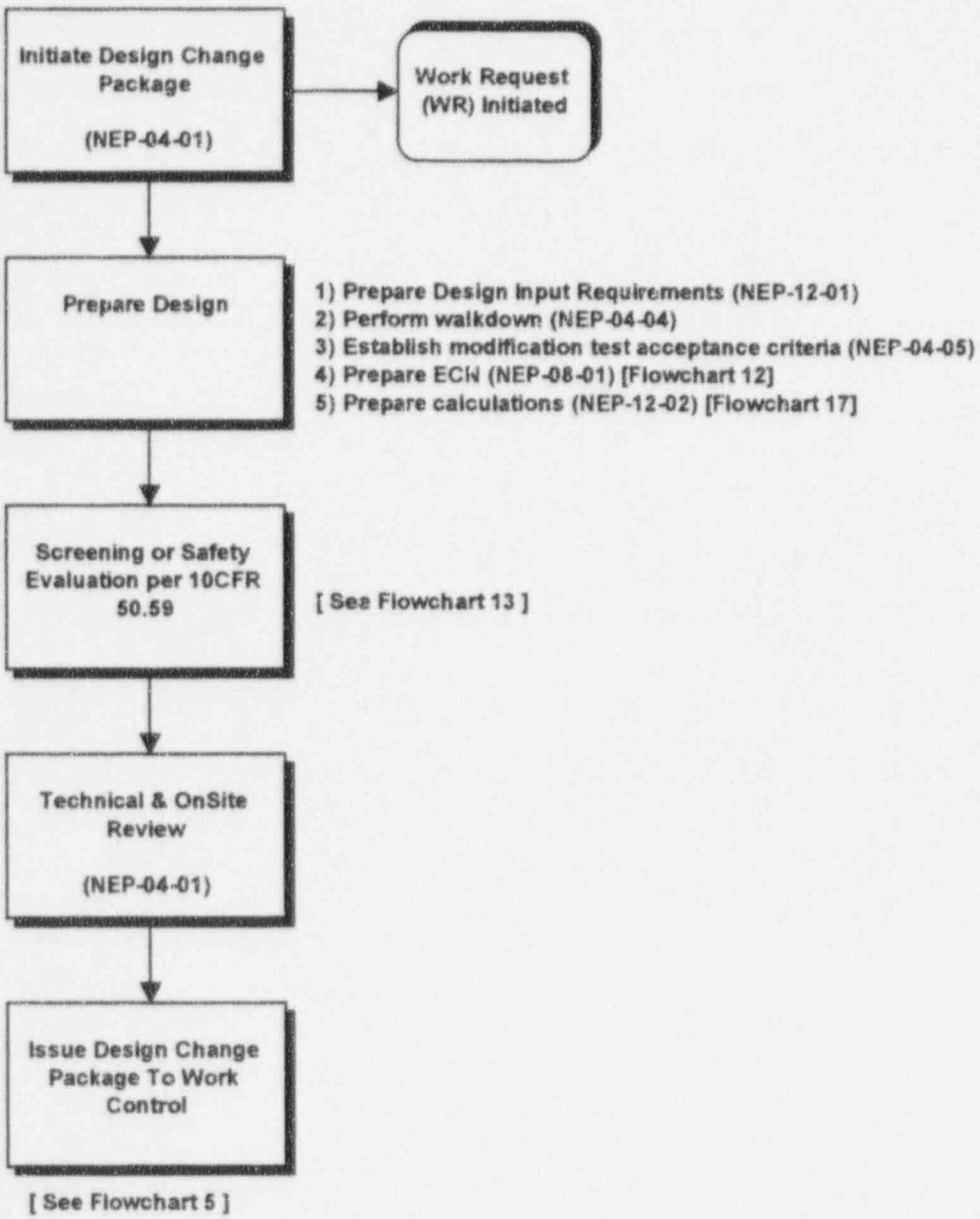
CHECKS AND BALANCES

Although there are areas within the process that provide overall reviews of the design, several specific areas provide for independent reviews against the design bases. The first area is handled through Engineering Change Notices (ECNs), which are used to develop the detailed design. Each ECN goes through an interfacing review process, an independent reviewer, and an approver. Similarly, engineering calculations are prepared to support the design indicated on ECNs and go through an interfacing review process, an independent reviewer, and an approver. A 10 CFR 50.59 safety evaluation is also part of the design process and provides an additional opportunity to review the changes against the design bases. The ECN, calculation and safety evaluation process are described separately in more detail.

Walkdowns performed after installation, as described in the Modification Work Control Process, also provide another area where the design is evaluated to ensure that it has met the original design requirements. If the design is found to be installed outside of tolerance or an alternate design configuration is required, a Field Change Request (FCR) is generated to evaluate the differences. All FCRs go through the same rigor of evaluation as the original design. Additional engineering calculations and 10 CFR 50.59 safety evaluations may be required.

Post Modification Testing, as discussed in the Modification Work Control Process, is the last area where the design is evaluated to ensure that it has met the original design requirements.

Flowchart 4
Engineering Design Change Process
NEP-04-01



Modification Work Control, Process 5

PURPOSE

The purpose of this process is to provide the necessary controls for the development of work packages which include installation instructions, quality control review expectations, and post modification testing requirements prior to Operations Authorization of the modification.

PROCESS DESCRIPTION

Once the Design Change Package (DCP) is issued, a Work Package is prepared that provides the necessary instructions for installation, Quality Control (QC) reviews, and testing. During the installation phase, a pre-installation walkdown is performed, Field Change Requests (FCRs) are generated for variations to installation requirements (if required), and post-installation walkdowns are performed to ensure that the modifications are installed per the construction documents.

After installation, a QC review is completed, post modification testing is performed, associated training is completed, and all configuration control issues are addressed. This includes updating Critical Control Room Drawings (CCRD) and operating procedures. Open items that are not needed for Operation Authorization, are identified and tracked separately for future closure.

The modification is then "Operations Authorized" and a "Turnover" is issued incorporating changes to the affected design documents.

CHECKS AND BALANCES

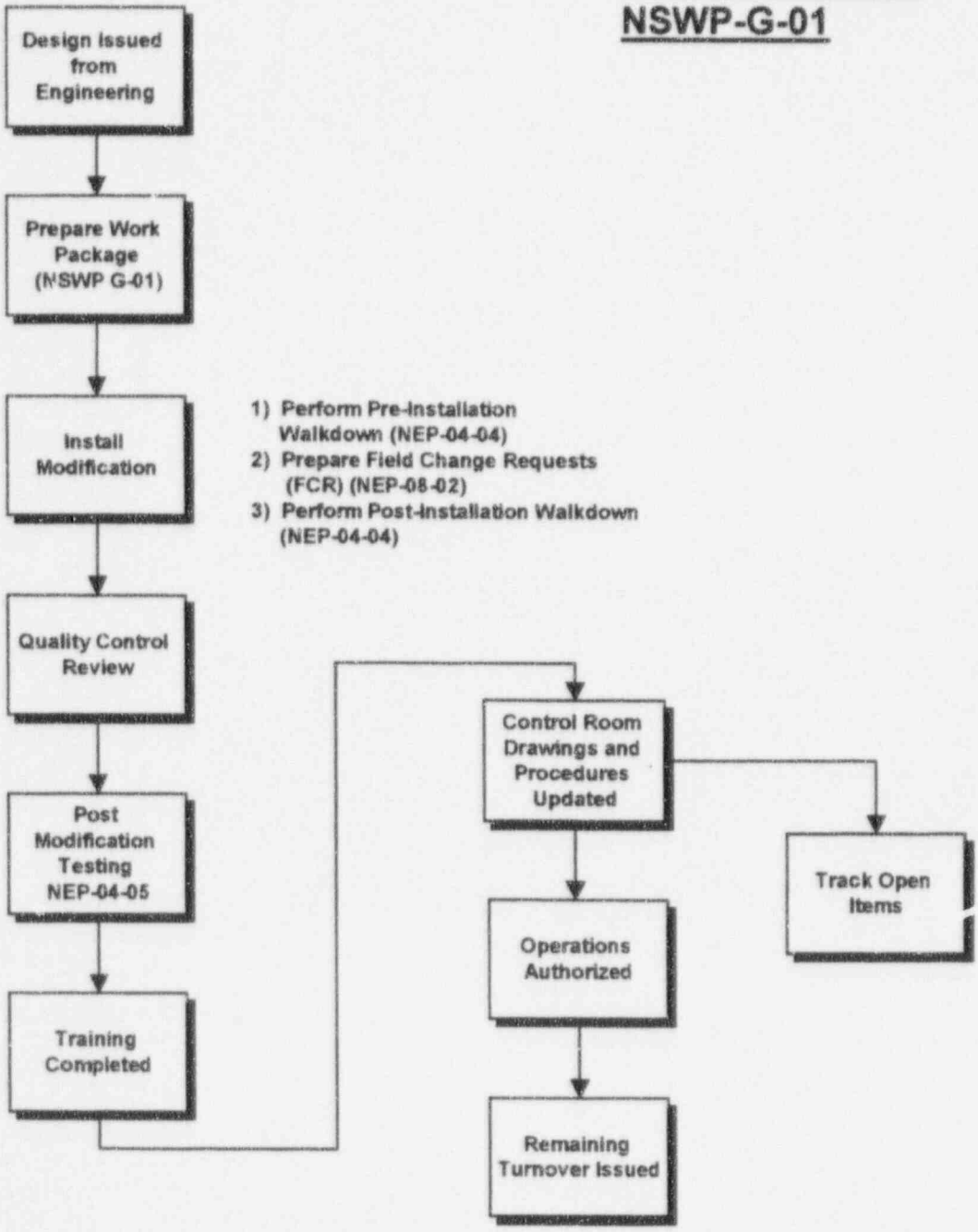
The pre-installation walkdowns provide an opportunity to evaluate the modification against the physical attributes and design considerations of other components located in same area. Changes required during this evaluation and others required during the installation, are all evaluated through the Field Change Requests (FCRs). FCRs take each deviation and evaluate it against the same criteria used for the original design. This includes independent reviews and 10 CFR 50.59 safety evaluations, if applicable.

Post-installation walkdowns and testing are performed to ensure that the modification is installed as designed and that it functions as intended.

RECENT/PLANNED IMPROVEMENTS

A Corporate-wide initiative is currently underway to improve "getting work done" within ComEd. This initiative includes the Work Control Process as an important element of the overall objective.

Flowchart 5
Modification Work Control Process
NSWP-G-01



Temporary Alterations (Temp Alts), Process 6

PURPOSE

The Temporary Alteration (Temp Alt) process is intended to provide assurance that a Temp Alt made to plant equipment does not degrade plant safety/reliability or unacceptably alter the approved design configuration.

PROCESS DESCRIPTION

The first step is to determine if the proposed change meets the definition of a Temp Alt. If not, the change must be processed using one of the permanent design change processes. If it does meet the definition, it can be processed as a Temp Alt or using an Administrative Controlled process that is specific to the type of change being considered.

With the Temp Alt process, design sketches and calculations are prepared, as required. When needed, walkdowns are performed and a 10 CFR 50.59 screening/safety evaluation is completed, as appropriate.

A Design Impact Evaluation and Modification Review Checklist are completed. The design goes through an independent review process and the Temp Alt is approved and issued.

CHECKS AND BALANCES

The first checkpoint involves the control to ensure that permanent changes are not processed as a Temp Alt. Permanent change processes are available that provide the appropriate level of controls. A 10 CFR 50.59 screening or safety evaluation is required for each Temp Alt. This process is described separately.

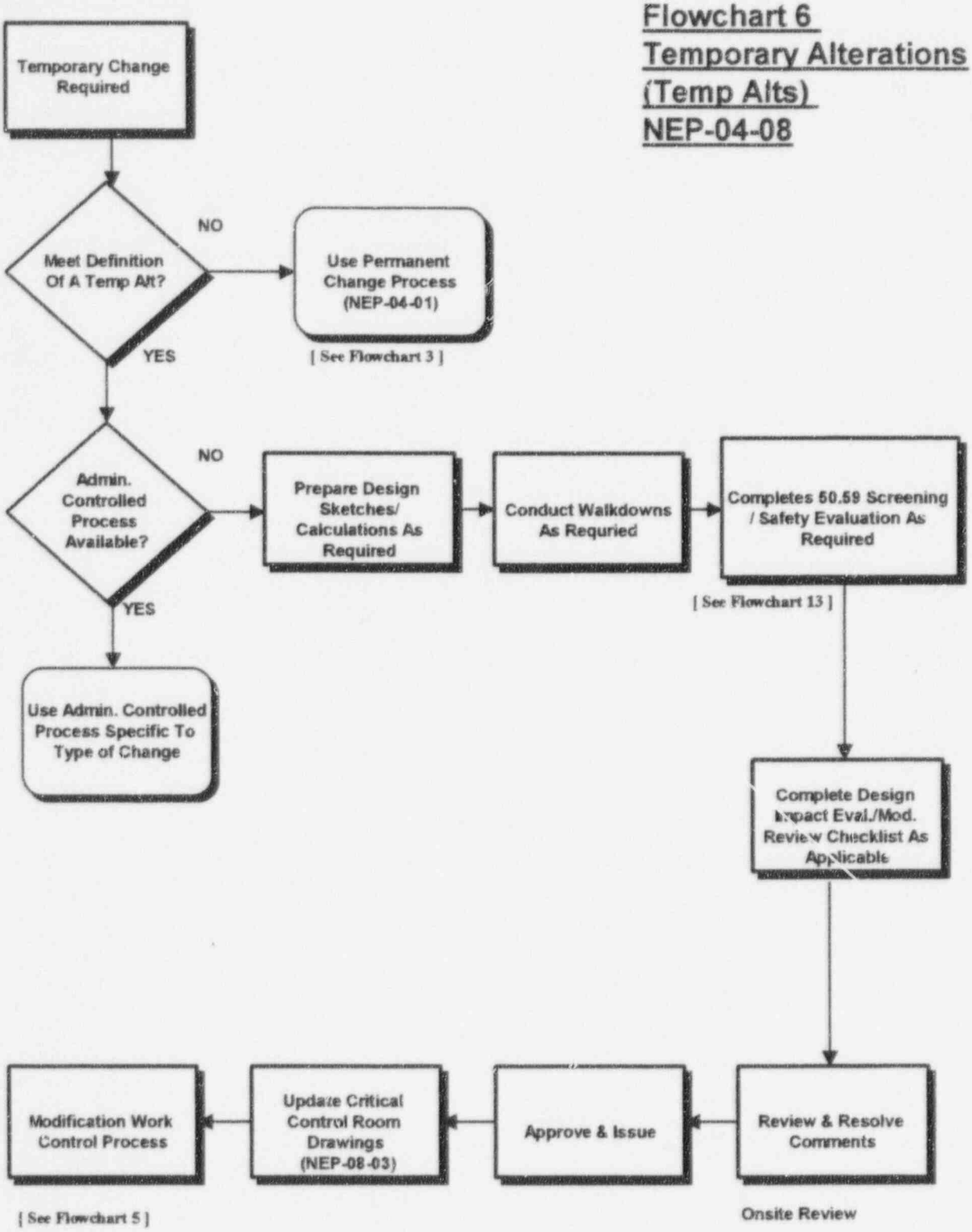
A Design Impact Evaluation/Modification Review Checklist is used to ensure that plant safety and reliability are not adversely affected, proper design control is maintained through a verification that appropriate drawings and procedures are revised to reflect the temporary configuration, and that testing considerations are addressed.

Temporary Alterations are required to be updated on the Critical Control Room Drawings (CCRD) so that these are maintained to reflect the plant configuration at all times.

RECENT/PLANNED IMPROVEMENTS

A six site evaluation team that has been formed to review Temp Alt issues that were identified through various reviews. This team has established root causes and proposed solutions that simplify the process, improve the understanding of what is considered a Temp Alt and standardize the process at all six sites.

Flowchart 6
Temporary Alterations (Temp Alts)
NEP-04-08



Document Change Requests (DCRs), Process 7

PURPOSE

The Document Change Request (DCR) process is used to control incorporation of design changes or as-built information into design documents. This document is initiated through the Electronic Work Control System (EWCS).

PROCESS DESCRIPTION

When a document change is required, two separate paths are provided depending on the source of the change. If the required change is the result of a Design Change, then an Affected Document List (ADL) is prepared and is reviewed against other pending changes. Engineering Change Notices (ECNs) and Field Change Requests (FCRs) are incorporated, and the documents are reviewed, approved, and issued.

If the required change is the result of an as-built condition, then an ADL is prepared, it is reviewed against other pending changes, and a 10 CFR 50.59 Screening/Safety Evaluation is prepared. If no Unreviewed Safety Question has been identified, the documents are updated, reviewed, approved, and issued.

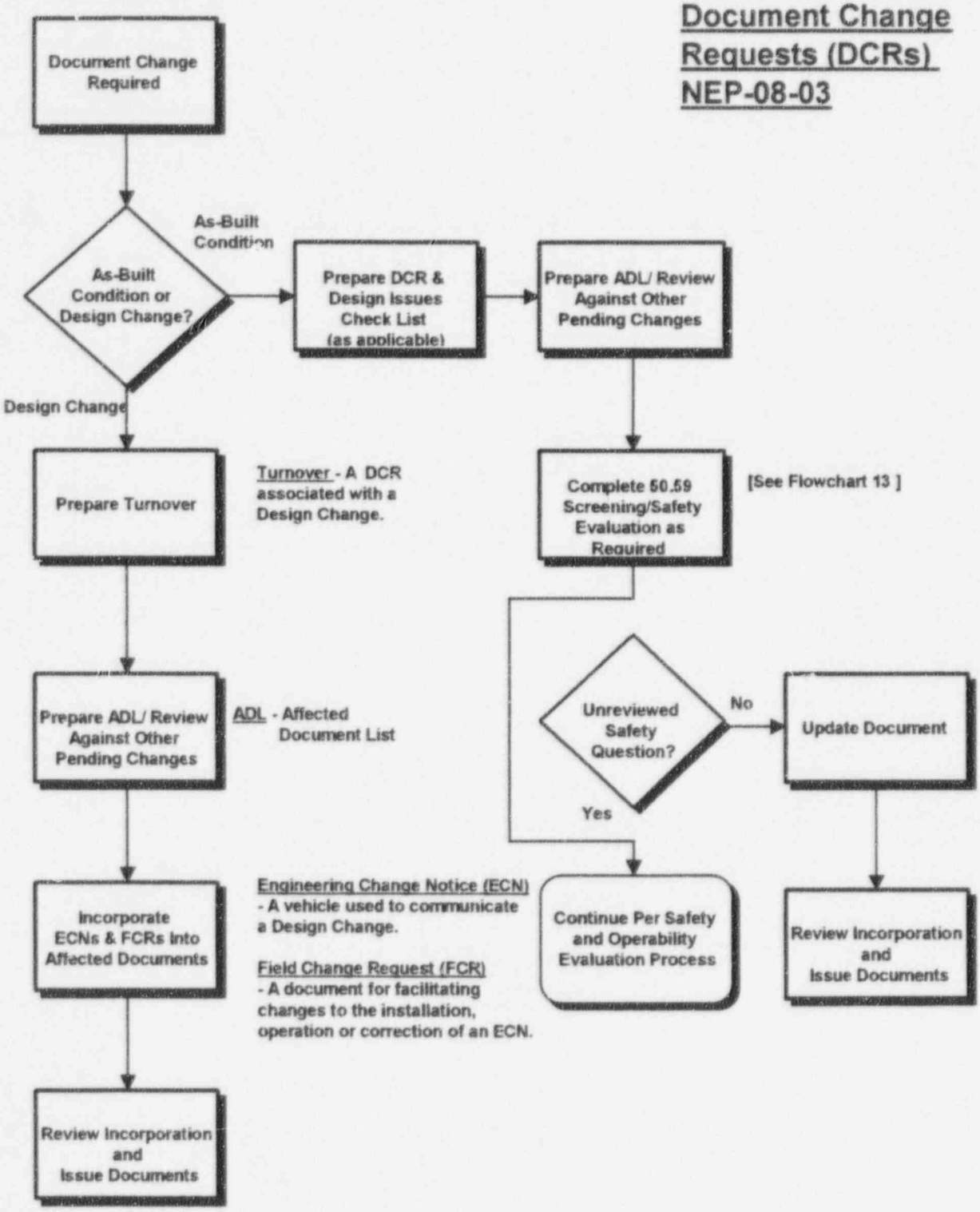
CHECKS AND BALANCES

There are several areas within this process that provide additional checks for reviewing the proposed change against other pending changes and design issues. Several of these checks are accomplished through the main elements of EWCS, which are described separately.

When preparing the ADL, EWCS is used to identify all outstanding changes that exist against the current revision of the document. This aids in determining the full impact of the proposed change for as-built evaluations and for combining information for document updates. A Turnover/DCR Design Issues Checklist is also provided for use in determining the impact of as-built changes in reference to several design issues.

The 10 CFR 50.59 Screening/Safety Evaluation process, which is described separately, is tied to processing all as-built changes. When a document and physical plant mismatch is discovered, a design engineer reviews the design to ensure it is physically correct before automatically assuming the documentation is incorrect from a design perspective.

Flowchart 7
Document Change Requests (DCRs)
NEP-08-03



Like-For-Like or Alternate Replacement Evaluation, Process 8

PURPOSE

The purpose of the Material Procurement Process is to establish uniform criteria for procurement of items and services that will be used for operations, maintenance, and modification of ComEd nuclear units with the following objectives:

- Ensure installed items comply with the plant Design Bases
- Ensure the configuration gets properly documented
- Minimize cost to the company
- Maximize the use of existing inventory
- Minimize inventory
- Minimize procurement effort
- Maximize the use of technically acceptable alternates

The company received recognition on the effectiveness of its program in August 1992 by an industry independent assessment group and conferred the title of Good Practice on the material procurement dedication processes.

The scope of the process includes new and replacement items for quality related applications. The process also describes the relationship between design, qualification, procurement, dedication, and supply.

PROCESS DESCRIPTION

Once the need for an item is identified, a determination is made whether an item has previously been identified for use in the specific application. If the answer is no, the design requirements for the item are established. The design requirements may apply to current design and/or those required for a design change. Design requirements are identified through: review of design document, equipment walkdown, safety classification data, technical data on form, fit and function, and design qualification documentation.

Should a replacement other than like-for-like [identical] design be required, the process directs the user to the correct procedures for continuation of the process depending on the complexity: Technical Evaluation [NEP-11-01], Alternate Replacement [NEP-11-01], or Modification [NEP-04-01]. The process includes a 10 CFR 50.59 Screening or Safety evaluation and independent engineering review and approval. When qualification of design is required for new or replacement items, the process directs the user to the appropriate design qualification methods.

Once the design, qualification and description of the items are completed, the process directs the establishment of requirements for the procurement of items through the supply process. Verification that items specified are those that are procured is through the Quality Receipt Inspection process.

The process requires the use of the following forms and checklists from NEP 11-01:

- Component Information Form-14
- Dedication Checklist Form-22
- Technical Evaluation Checklist Form-23
- Alternate Replacement Checklist Form-24

The checklists contain reference to design and license documents. They are derived from the following EPRI Guidelines.

> EPRI NP-5652, "Guidelines for the Utilization of Commercial Grade Items in Nuclear Safety-Related Applications [NCIG-07]"
>
> EPRI NP-6406, "Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants [NCIG-11]"

CHECKS AND BALANCES

A number of checks and balances exist in the current process. Safety-related material purchase orders are quality records and provide a link to the original equipment design specifications. The technical and quality requirements imposed on the purchase of material that reflect the design of the item are a result of the material engineering procedures NEP 11-01. The process requires an independent engineer review and approval of completed work. The verification that purchase order requirements have been met is accomplished through a combination of receipt inspections, dedication testing and engineering review of test results. The receipt process includes independent quality control overview. ASME code items undergo additional verification by Authorized Nuclear Inspectors with the process periodically audited to ASME 626 criteria.

The process is audited annually by ComEd Quality Verification to the appropriate requirements of 10 CFR 50 Appendix B. Corrective actions are identified and program revisions are made. The process has undergone independent review and self-assessment a number of times since 1990 with corrective actions made based on the weaknesses identified.

Strengths include:

> A process and program recognized by industry peer evaluation as a Best Practice supported by standardized procedures, and significant resource with state of the art inspection and testing tools.

The process includes reverse engineering criteria, which has evolved for similar applications in military and aerospace programs where maintaining design of items are critical and a suitable replacement is available in the supply chain.

Weaknesses include:

Prior to 1990, procedures governing the process were not standardized across the six stations. Common problems existed. Fraudulent material concerns were noted by the NRC in 1988.

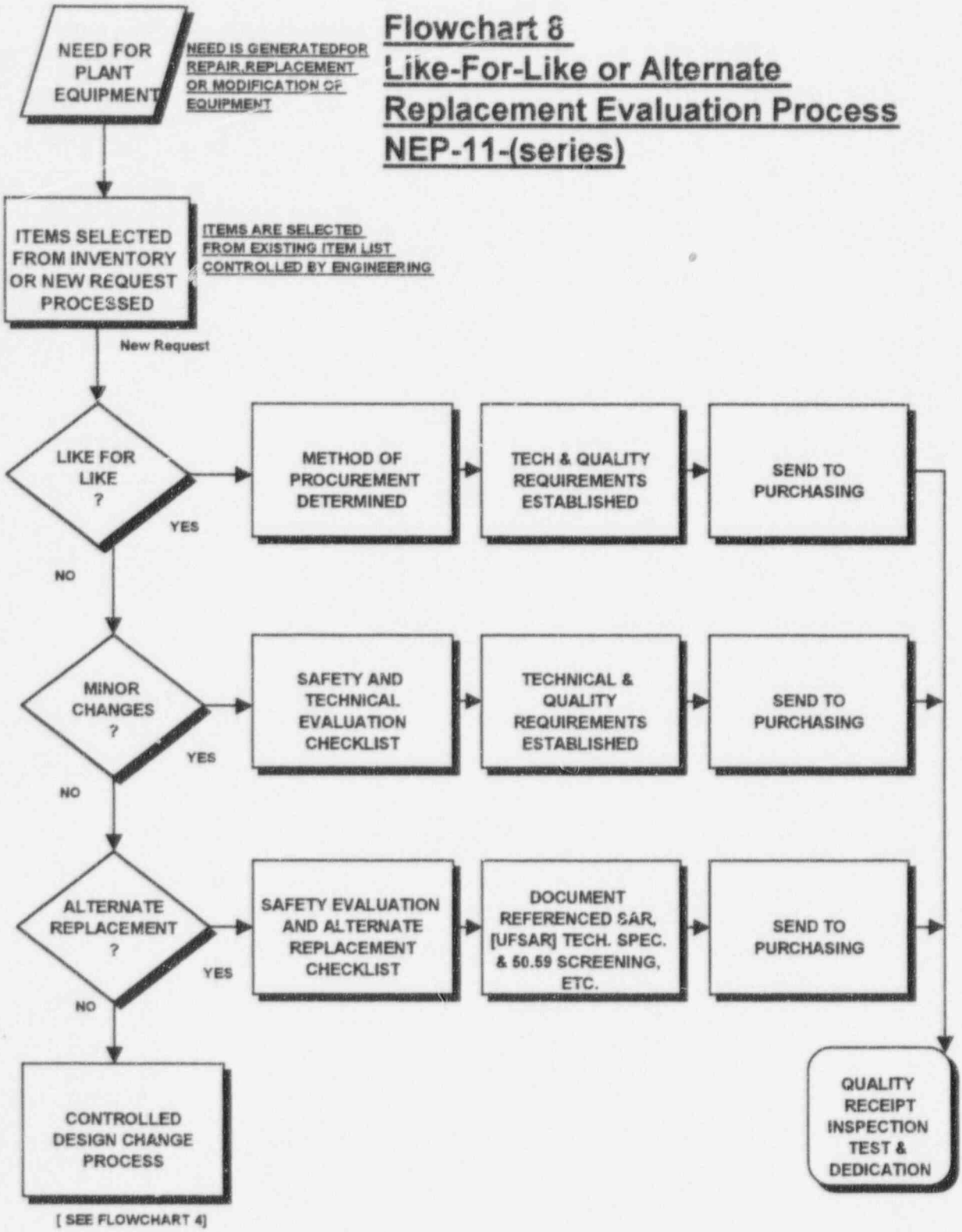
Application of parts engineering procedures, and process was mandatory for safety-related and regulatory related equipment only. Use of procedures and process was optional for non safety equipment.

RECENT/PLANNED IMPROVEMENTS

Corrective actions for current program weaknesses have been established. Implementation of current corrective actions began in October 1996. Parts Engineering procedures are applicable to systems and components referenced in the plant's UFSAR.

Qualification and training of parts engineers was originally under site specific programs. Current training of parts engineers is accomplished through a combination of EPRI sponsored and managed programs combined with ComEd specific criteria. The program contains two levels of qualification. The training process has been revised to include ACAD 91-017 criteria.

Flowchart 8
Like-For-Like or Alternate Replacement Evaluation Process
NEP-11-(series)



Setpoint Change Request, Process 9

PURPOSE

The goal of this process is to establish a standardized, computerized Instrument Database, with supporting documentation and a single point of control, implemented consistently at all six stations.

PROCESS DESCRIPTION

The Requester completes the initiation section of the station Setpoint Change Request Form.

Engineering Supervisor reviews the Setpoint Change Request (SCR) to validate the safety classification, to recommend training and procedural changes, and to determine whether a modification is required. If a modification is not required, the SCR is forwarded to Operations.

Operations reviews the Setpoint Change Request to determine system operating impact, and forwards it to Training.

Training reviews the Setpoint Change Request to validate/recommend the training requirements and then returns it to Engineering.

If a modification is required or the Setpoint Change Request is classified as safety or regulatory related, Engineering performs a technical review and approval.

The Engineering Technical Review shall address the following items:

1. Performance of a 10 CFR 50.59 screening or safety evaluation.
2. Determination of a need for Nuclear Fuel Services (NFS) review. If the change affects Reactor Protection and Control setpoints or a setpoint used as an input to the safety analysis, NFS must be notified.
3. Confirmation of compliance with applicable regulatory guidelines and Industry Standards.
4. Performance of a document review to ensure that the proposed Setpoint Change is in accordance with the design bases.
5. Confirmation of recommended training or recommending additional training.
6. Identification of QA/QC related items and audit or inspection points.
7. Completion of human factors review, as applicable.

The setpoint change and testing is implemented per the appropriate station procedures. Close-out of a Setpoint Change Request is accomplished in accordance with the Setpoint and Data Change Request, and Document Change Request Procedures. A DCR is then initiated to update the appropriate design documents and/or data-bases.

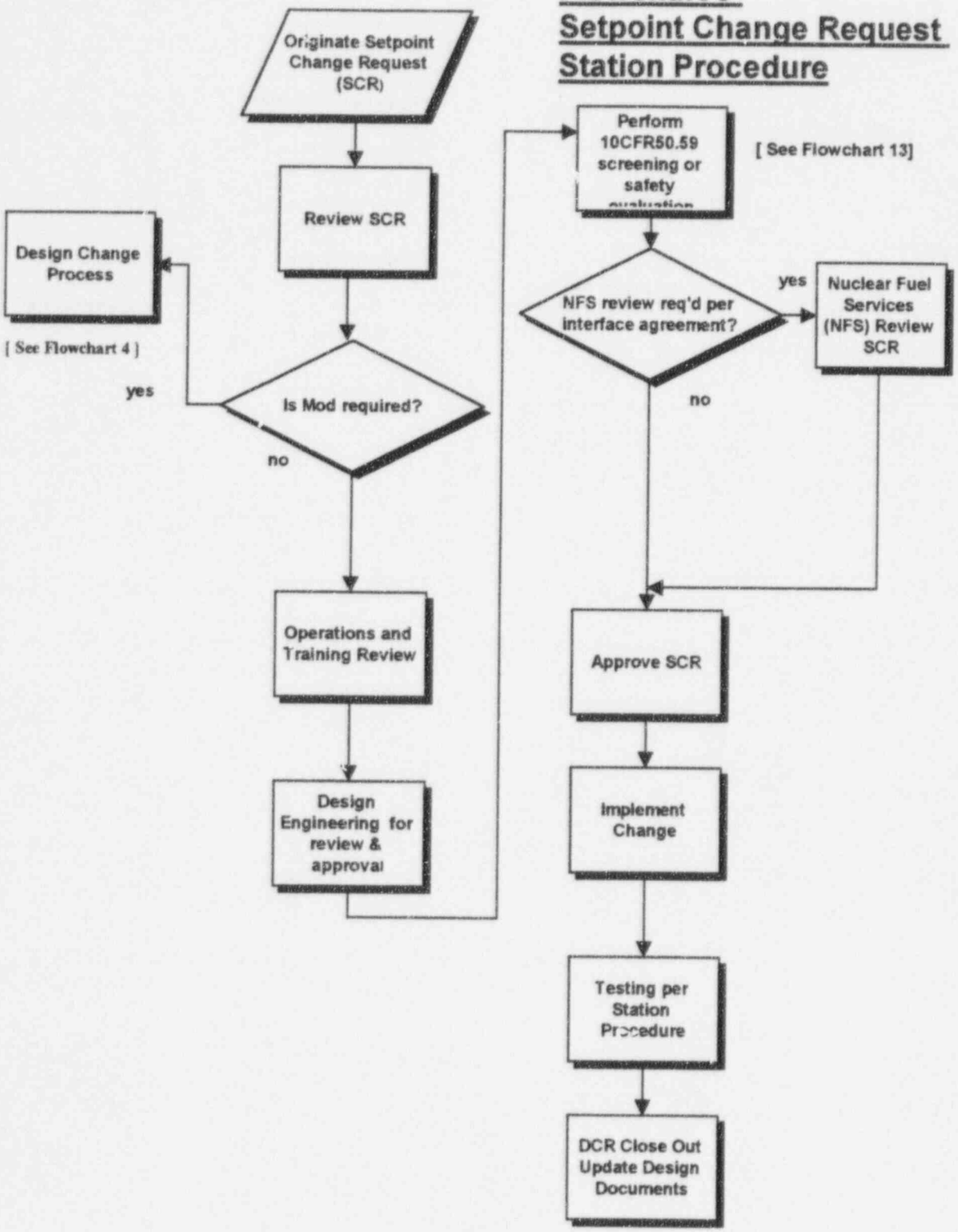
CHECKS AND BALANCES

The independent review performed by Operations and Training to determine operations and training impact of the setpoint change offers an early station perspective in the process to ensure the change is correctly processed and the impact is fully understood.

RECENT / PLANNED IMPROVEMENTS

The Instrument Database was developed to control instrument design and maintenance information. Input from various AE's, station departments and engineering was used to develop a standardized database, common to all six stations. Various methods of initial data entry were utilized, ranging from transfer of existing Instrument Index and Data Sheet information (Byron, Braidwood, Zion) to complete plant walkdowns and document reviews (Dresden, Quad Cities, LaSalle). Procedures are being written and implemented at each station and within Nuclear Engineering to control and populate the database.

Flowchart 9
Setpoint Change Request Station Procedure



OMITTED - NOT USED AT BYRON

BYRON HAS NO DBDs

Engineering Software Development and Revision, Process 11

PURPOSE

The Engineering Software Program applies to software that is safety-related, used to perform controlled work, used to verify Station Technical Specification compliance or used to comply with regulatory requirements not contained in the Technical Specification. This process specifically describes the steps used to control revisions to Engineering Software.

PROCESS DESCRIPTION

Once a need to develop or revise Engineering Software has been identified, a Software Activity Request is filled out to describe the situation and identify the activities that need to be performed.

A Software Management Plan (SMP) is generated that includes:

- identification of the Software Product.
- responsibilities and schedules.
- required documentation.
- standard, conventions, techniques or methodologies
- compatibility with other engineering computer programs.
- required reviews.
- error reporting method.

A Software Requirements Specification (SRS) is then developed to describe:

- the functions the software is to perform.
- the software performance.
- design constraints.
- attributes.
- external interfaces.

The programming change will then begin based on the documents generated above, in preparation of software testing. A preliminary test case shall be used to validate the Engineering Computer Program (ECP) to assure that the software produces correct results for the test case.

CHECKS AND BALANCES

Software Verification and Validation (SVV) activities shall begin with the development of a SVV Plan (SVVP) which describes:

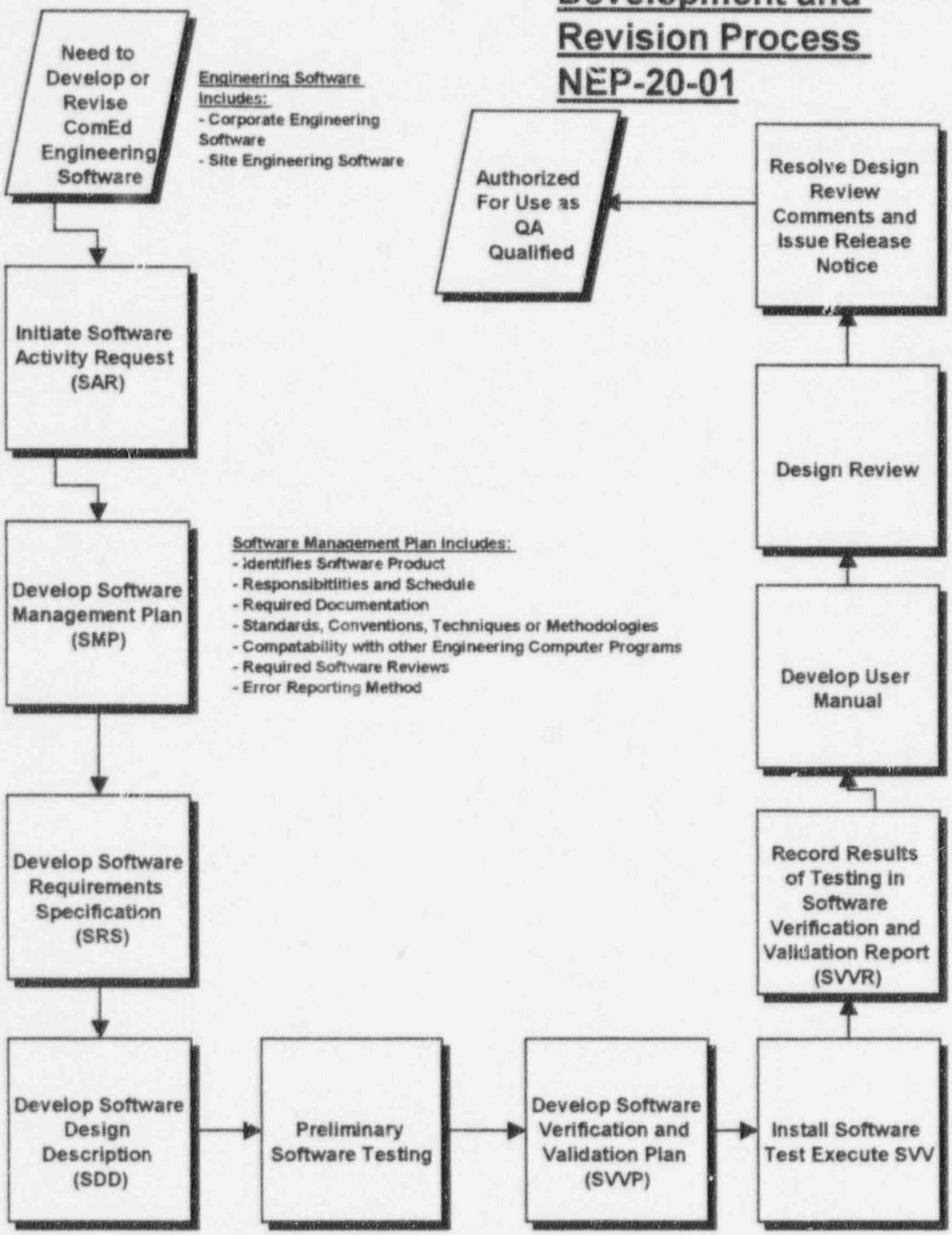
- tasks and criteria for accomplishing the Verification of the ECP.
- hardware and software configurations pertinent to validation and verification.
- tracability to both the software requirements and the software design.

The software is then installed, tested and the results documented for review in a Software Verification and Validation Report. A user manual is then prepared for review.

A Design Review, as defined in NEP-20-01, is required prior to designating the software as qualified for controlled work. This review ensures that the requirements of the engineering software have been fully met and documented.

The results of the Design Review are documented through a release notice and the software is authorized for use.

Flowchart 11
Engineering Software Development and Revision Process
NEP-20-01



Engineering Change Notices (ECNs), Process 12

PURPOSE

Engineering Change Notices (ECNs) are used to communicate design changes which are included in a Design Change Package. They are initiated through the Electronic Work Control System (EWCS) and provide for a systematic approach to support the preparation, review and approval on the ECN.

PROCESS DESCRIPTION

Once the ECN is initiated, all affected documents and required calculations are identified on the Affected Documents List (ADL). Initial configuration changes/additions are prepared and pre-installation plant walkdowns are performed, as required. Detailed designs and engineering calculations are then prepared and a package is distributed for interfacing comments.

After interfacing comments have been resolved, the ECN goes through an independent review process, and is then approved and ready to be included in the Design Change Package for forwarding to the Modification Work Control Process.

CHECKS AND BALANCES

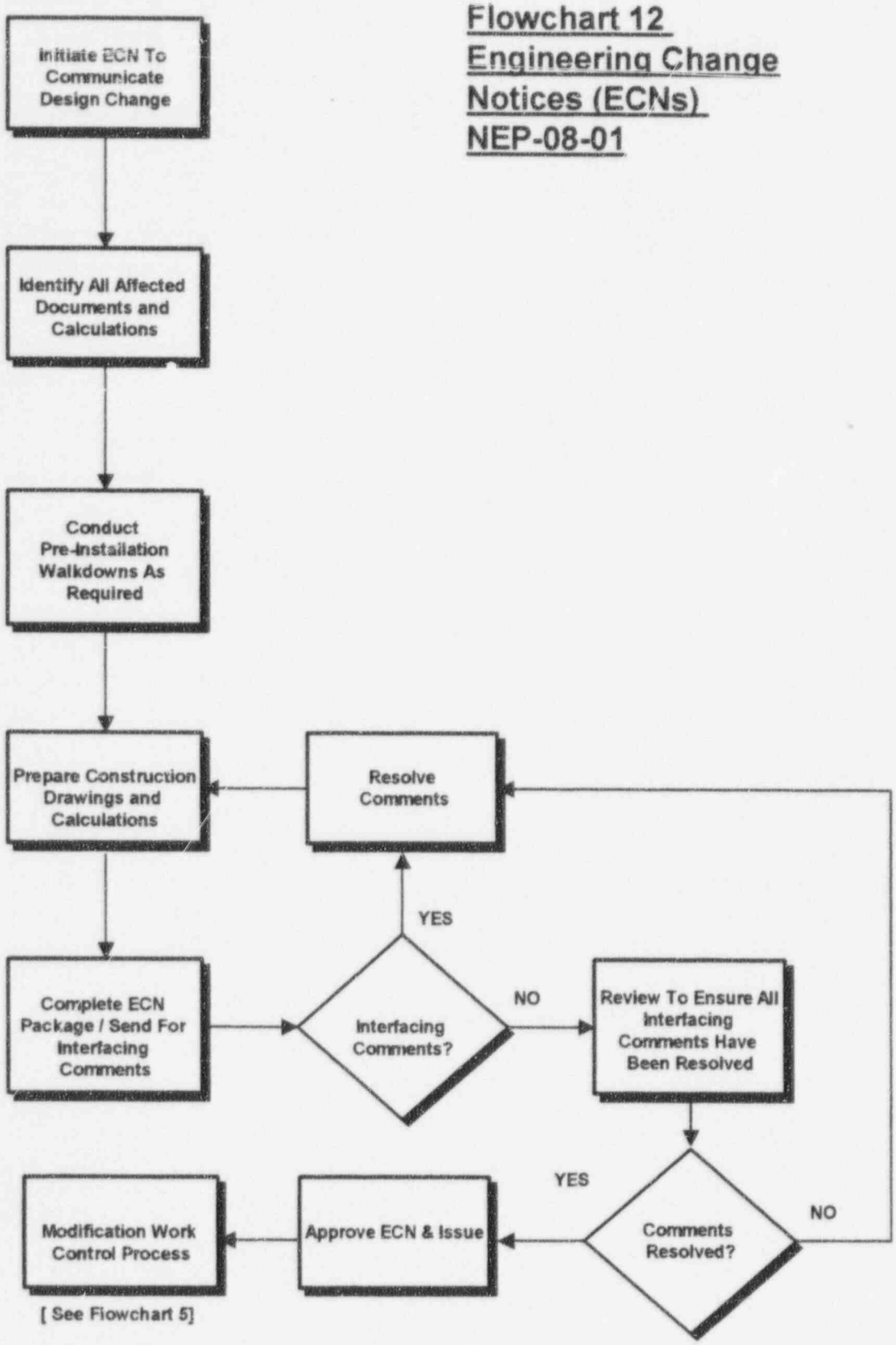
As the ADL is prepared through EWCS, all pending changes are identified and evaluated for their impact to the new change/addition. This allows for an additional evaluation of all previously planned changes and those which are currently underway.

The interfacing comment step provides for a technical evaluation in specific related areas that interface with the all aspects of the design. The evaluation is performed by those with expertise in the specific areas and are performed independently.

RECENT/PLANNED IMPROVEMENTS

The list of potentially affected design documents to be included in the ADL was recently revised to provide more detailed guidance to the preparer. This should improve the accuracy of the initial ADL.

Flowchart 12
Engineering Change Notices (ECNs)
NEP-08-01



Safety Evaluation, Process 13

PURPOSE

The purpose is to determine and provide a documented basis for concluding if an Unreviewed Safety Question exists for a change, test, or experiment, or if a technical specification change is required.

PROCESS DESCRIPTION

Reviewers and preparers must be trained and qualified to perform Screenings and Safety Evaluations.

A Screening is performed to determine whether a safety evaluation is required per 10 CFR 50.59. The safety evaluation provides the review to determine whether an unreviewed safety question or technical specification change is involved.

The Preparer reviews the UFSAR, pending UFSAR changes, and other Licensing Basis documents and describes how the proposed activity will affect plant operations or potential equipment failures.

The Preparer identifies activities that could be affected by the change and determines if new or revised Technical Specifications are needed.

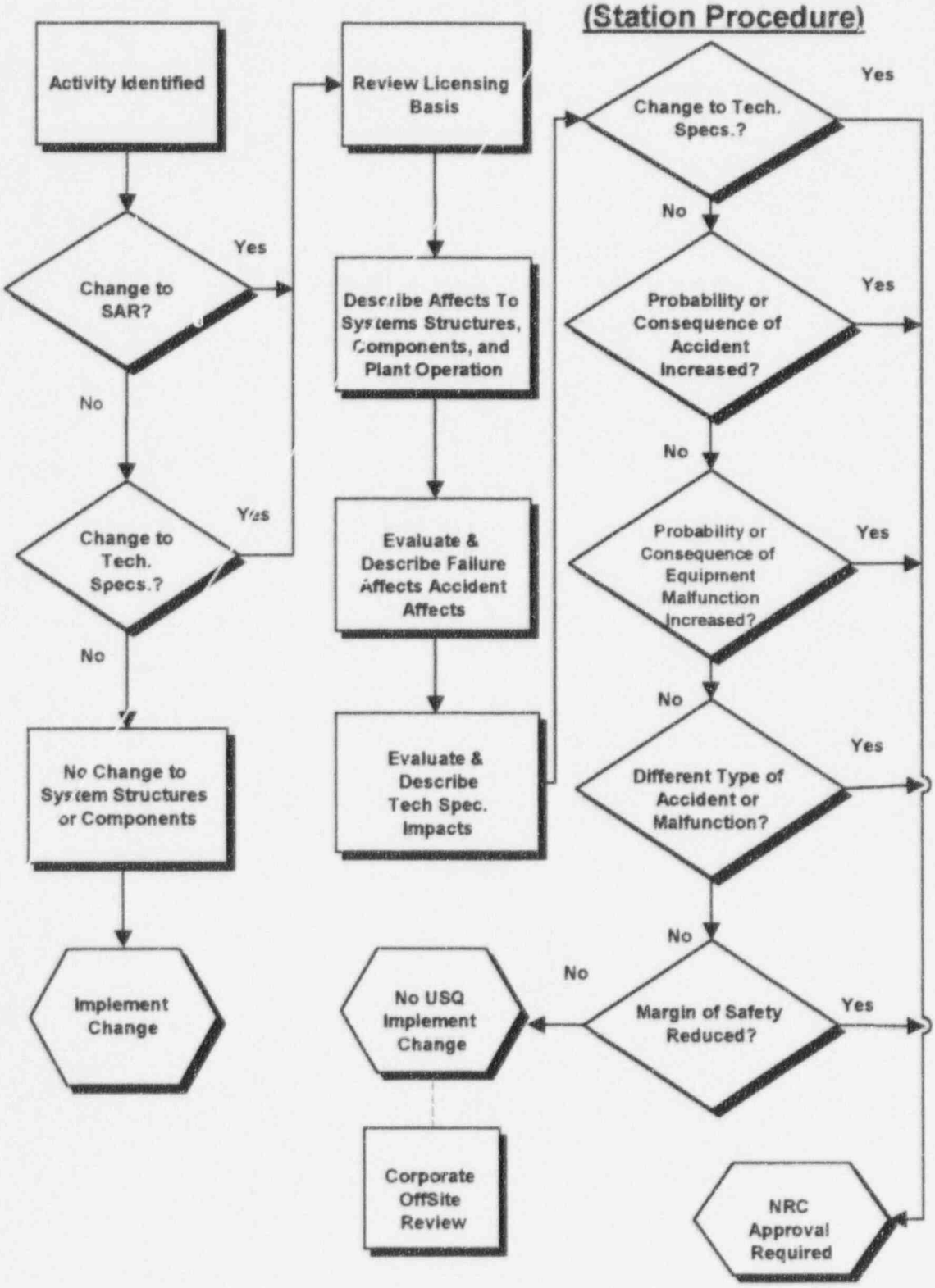
Selected completed 10 CFR 50 59 reviews are independently reviewed by the Off Site Review Group for the following:

- No Unreviewed Safety Question is involved
- All questions are properly answered
- The supporting documentation justifies conclusion
- Technical Specification change not needed

RECENT/PLANNED IMPROVEMENTS

ComEd's Safety Evaluation Process has been the subject of several NRC Audits. The specific findings and ComEd's corrective actions are discussed in the station attachments. ComEd is in the process of adopting a uniform safety evaluation process.

Flowchart 13
Safety Evaluation Process
(Station Procedure)



VETIP, Process 14

PURPOSE

This process provides a methodology for the control of vendor technical information used for the installation, maintenance, operation, testing, calibration, troubleshooting, and storage of equipment. In compliance with ComEd's commitment to NRC Generic Letters 83-28 and 90-03, all vendors supplying critical safety-related components are recontacted periodically (at least every three years) to ensure the latest manual revision is in the VETIP system.

PROCESS DESCRIPTION

All vendor manual information will be received and processed through the VETIP Coordinator at the station. The following activities will be performed for each vendor manual:

The coordinator reviews the manual for applicability and to determine whether the manual is currently in use at the station.

If the new manual is a revision to an existing manual, the coordinator determines whether the change is administrative or technical and prepares a Vendor Document Comparison Report (VDCR) which summarizes the changes between the different revisions of the manual.

The VDCR and manual are forwarded to the subject matter expert (SME) for review. If the SME finds the changes acceptable, then the SME approves the manual and determines what other station groups should be notified of the manual changes. If station procedures are affected, the manual is forwarded to the procedure coordinator for incorporation as appropriate.

If the SME or other reviewers determine that the technical changes in the manual are not acceptable, the vendor manual is returned to the VETIP Coordinator for final disposition.

After review and approval by the SME, the VETIP coordinator updates other existing hard copies of the manual and updates databases. The original vendor information and all station review/approval documents are forwarded to Central Files for retention.

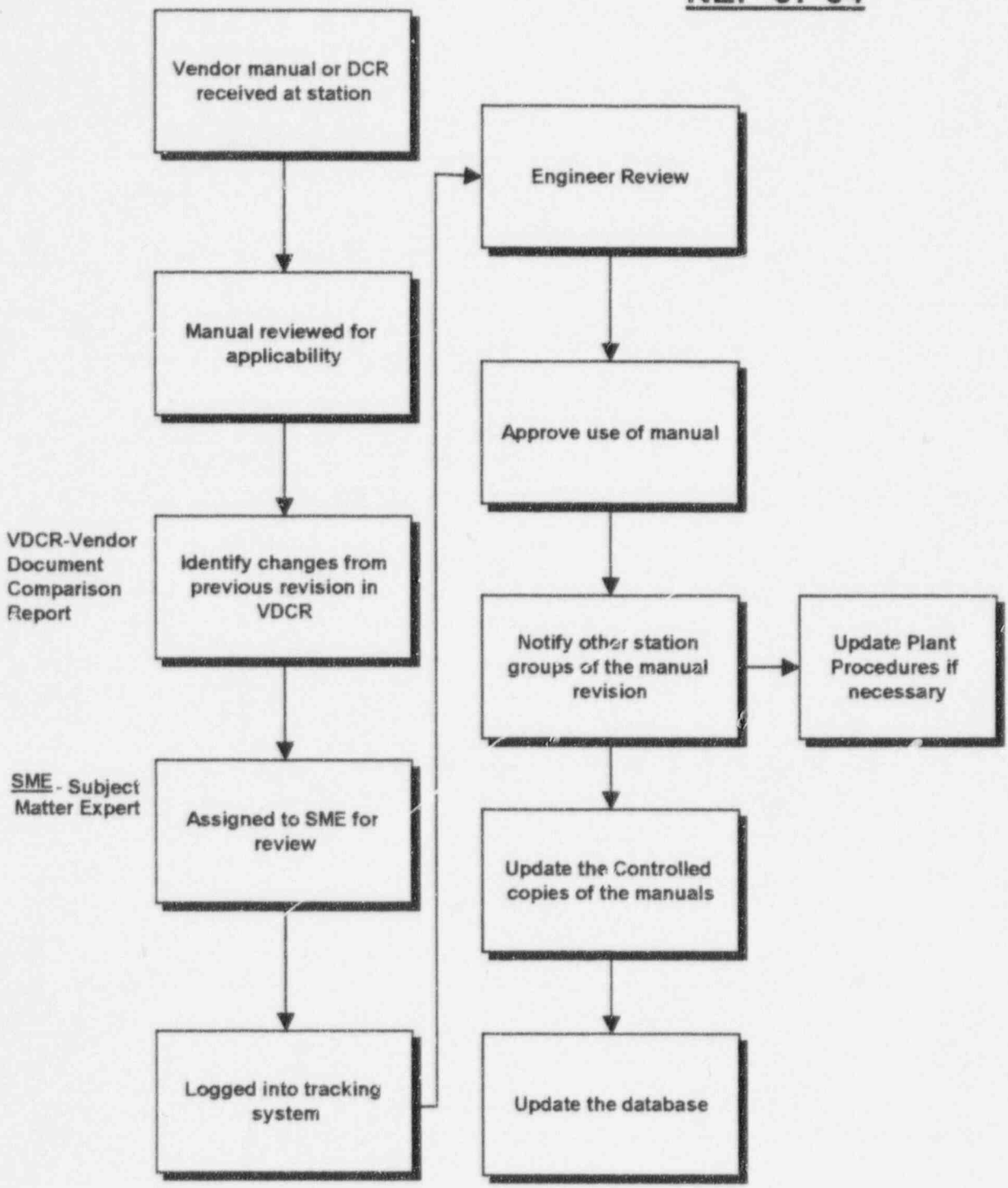
CHECKS AND BALANCES

The Subject Matter Expert Review concept ensures the right person is reviewing the manual in other cognizant departments.

RECENT/PLANNED IMPROVEMENTS

The process for changing the vendor manuals to the current status based on incoming OPEX documents, is not well proceduralized. The procedure governing VETIP is being revised by the VETIP Coordinators peer group to account for those changes.

Flowchart 14
VETIP Processing
NEP-07-04



Configuration Control Using EWCS, Process 15

PURPOSE

The Electronic Work Control System (EWCS) is an on-line workflow and database tool used at all six ComEd nuclear sites and the corporate offices. The elements of EWCS that are used to support configuration control are:

- Engineering Design Change Module (EDCM)
- Revision Tracking and Control
- Controlled Documents (CD)
- Equipment Database

These modules and their configuration control functions are outlined below.

PROCESS DESCRIPTION

Engineering Design Change Module

This module provides for assignment and status monitoring of 5 types of change documents. These are:

Engineering Requests (ERs) - Used to solicit assistance from engineering. ERs which may be closed by issuing a design change (only a small fraction of ERs become design changes) can be used to track the status of the change through the business review and technical review process.

Design Change Packages (DCPs) - Used as the over all tracking package for a collection of other change documents (DCNs, FCRs) or as the primary package for minor changes. When used for minor changes (simple, non-safety-related), DCPs require an Affected Document List (ADL) and Affected Equipment List (AEL) to track the status of impacted controlled documents and equipment data records through the change process.

Design Change Notices (DCNs) - Primary vehicle for issuing and tracking design changes. DCNs use ADLs and AELs to identify and track the status of impacted documents and equipment data records through the change process. DCNs must be associated with an overall DCP.

Field Change Requests (FCRs) - Used to issue and status field requested changes to support installation of issued DCPs. FCRs use ADLs and AELs to identify and track the status of impacted documents and equipment data records through the change process. FCRs must be associated with an overall DCP.

Document Change Requests (DCRs) - Used to document as found changes and discrepancies to design documents. DCRs use ADLs and AELs to identify and track the status of impacted documents and equipment data records through the change process. Note that a Turnover, not a DCR, is the vehicle used to track closure of document and equipment data changes associated with DCPs and DCNs and is part of those respective processes.

EDCM is the primary tool for tracking design and document changes from request to closure. Design interaction is readily identified through the use of the ADL and AEL.

Revision Tracking & Control (RT&C)

RT&C is technically a part of EDCM since it is initiated from the AEL. RT&C provides the ability to change equipment data associated with an EDCM change object through an on-line process. Anyone in the plant can initiate a data change request with this process. RT&C creates a temporary revision of each data record flagged as affected and allows this temporary change to be prepared, reviewed and approved on-line. When the design change is installed in the plant, the approved temporary revision is electronically issued into the EWCS equipment database.

Controlled Documents (CD)

CD is used as the controlled index to important plant document including drawings, calculations, procedures, and vendor information. The search features of CD are used by engineers and others to find and retrieve (from central files or through on-line viewing for some types of documents) these documents.

Equipment Database

The Equipment Database in EWCS is a common database used by engineering, maintenance and operations at each site. Users can search this database for equipment data such as safety classification, ASME code class, or electrical class. This data feeds into the on-line maintenance work requests and out-of-service requests to control quality requirements. Engineering controls critical equipment data in this database using RT&C. Multiple legacy databases are being migrated into this database to provide access to data for:

- Master Equipment List/ Quality List Data
- Valve Data
- Instrument Data
- Fuse Data

The Approved Model List is also an available feature of this database which can be used to effectively communicate evaluated alternate replacement components for a given application to maintenance. The Bill of Material feature is beginning to be used to provide detailed parts list for equipment in the system to greatly facilitate maintenance activities.

CHECKS AND BALANCES

When a document is identified as affected by the change and is placed on the ADL, Engineering Design Change Module (EDCM) searches the document database for other open changes against the document and immediately notifies the user if found. This feature is also in place for equipment records placed on the AEL.

Revision Tracking and Control (RT&C) also notifies all users of the EWCS equipment database when pending changes exist against the data they are viewing.

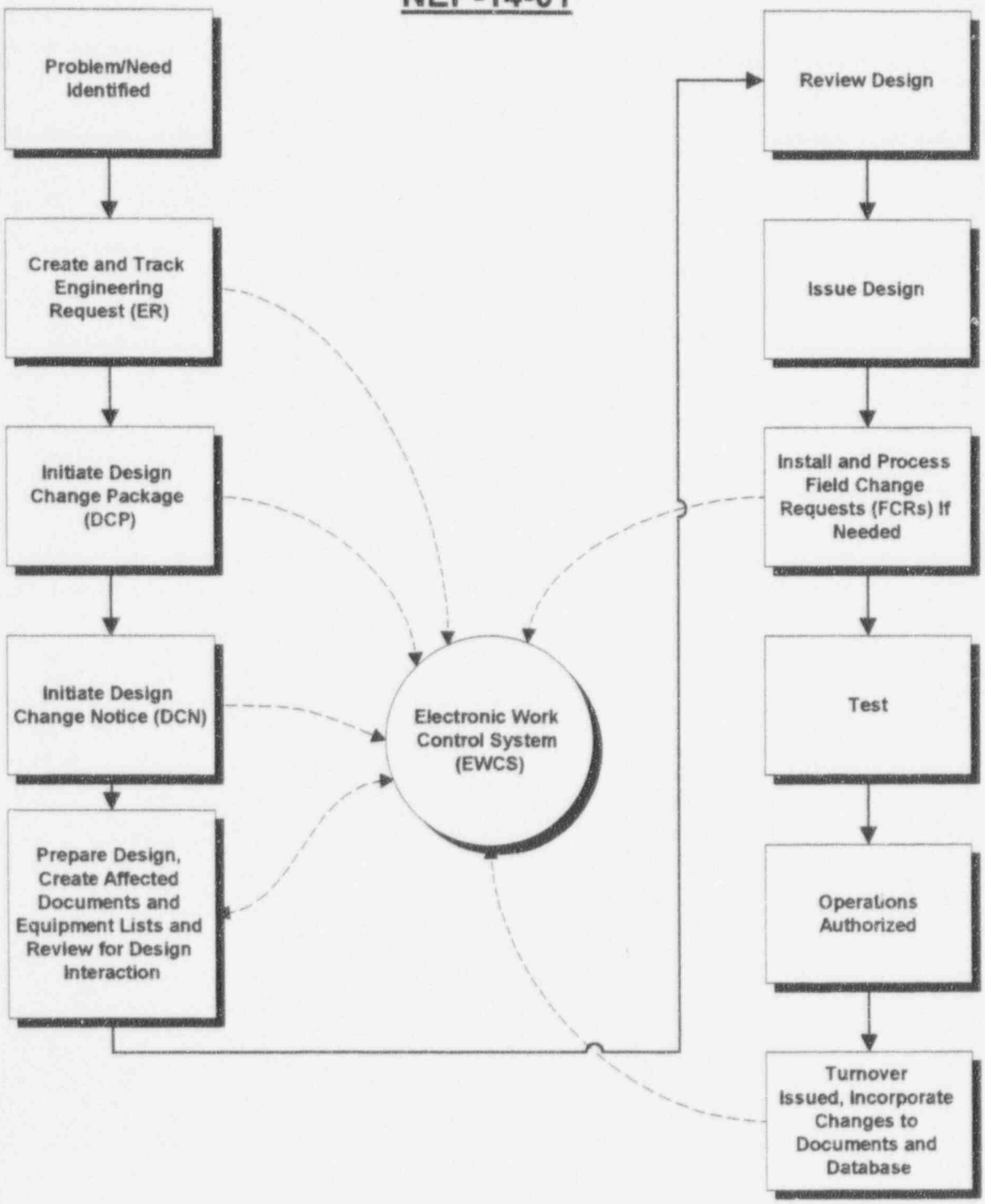
Like RT&C, Controlled Documents (CD) readily identifies to the user when outstanding changes exist against the current revision of a document. When a document has been checked out for use in the field, CD automatically notifies the user when a new revision is issued.

RECENT/PLANNED IMPROVEMENTS

Various legacy databases from AE and ComEd records are being entered into the Equipment Database in order to provide access to data associated with equipment lists, valve lists, instrument data and fuse data.

In addition, the Bill of Material feature of the Equipment Database is beginning to be used to provide detailed parts lists for equipment. This is expected to improve the consistency and significantly decrease the level of effort required to generate a Bill of Material.

Flowchart 15
Configuration Control Using EWCS
NEP-14-01



OMITTED - NOT USED AT BYRON

BYRON HAS NO DBDs

Calculation, Process 17

PURPOSE

This process describes the preparation, review, and approval requirements for calculations that support Engineering Design and Analysis.

PROCESS DESCRIPTION

The scope and approach to the calculation shall be established and applied.

Preparers are responsible for compiling the information and preparing the calculation in a prescribed manner for the stated purpose. Preparers shall possess discipline qualifications related to the subject matter or a specialization in the area through work experience, education, training, etc. During preparation, the Preparer shall be aware of the following which directly relate to the calculation:

Project files	Drawings
Meeting notes	Codes
Design criteria	Standards
Applicable previous calculations	Studies
System descriptions	Commitments to Regulatory Agencies

The preparer should adequately document Engineering Judgment, if applicable, to permit Reviewer to verify logic. Once the calculation is completed, the calculation may be checked prior to being submitted for an independent review. After all comments generated through the independent review have been resolved, the calculation is approved and issued.

CHECKS AND BALANCES

The Supervisor/Approver may check the calculation prior to formal review for:

Format	Attributes
Completeness	Reasonableness of results
Technical adequacy	

An "Independent Review" of Calculations is performed by a qualified individual, using detailed guidance, assigned by the Supervisor. The Reviewer shall have had no influence on inputs or approaches utilized in the design development. The Reviewer is responsible to ensure the calculations:

Are complete	Meet applicable codes

Are technically adequate	Meet applicable standards
Are accurate	Meet quality requirements
Are appropriate for stated purpose	Meet licensing commitments
Have appropriate of assumptions	Have reasonable output data

Calculations are reviewed by one or more of the following methods:

Detailed Design Review Method

Review calculations against design input documents to verify:

Conformance with specified configurations

Dimensions

Materials

Correctness of input parameters

Alternate Calculation Method

After ensuring that assumptions are appropriate and mathematics, input data or other calculation methods are correct, a simplified or approximate method of calculation is performed.

Qualification Testing Method

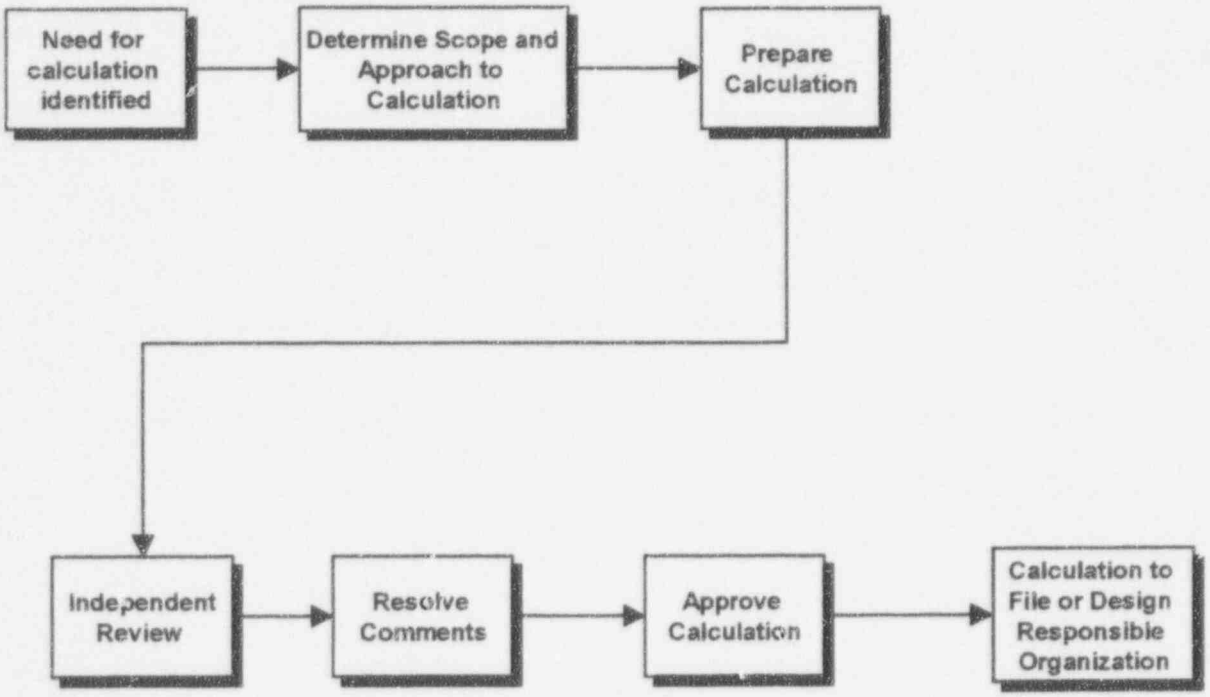
Verifying the adequacy of the calculation via a test program which demonstrates adequate performance under the most adverse operating conditions.

Review of Repetitive Calculations

Review previously approved calculations in terms of purpose, methodology, assumptions, and design inputs. Verify that differences will not affect the comparison and that conclusions are consistent.

Calculations are approved by the Supervisor or an experienced individual designated by the Supervisor. The Approver is responsible for the overall quality of the calculation.

Flowchart 17
Calculation Process
NEP-12-02



Operability Determination, Process 18

PURPOSE

Operability determinations are performed when the capability of a system, structure, or component (SSC) to perform its specified function(s) as required by the Technical Specifications or UFSAR cannot be unequivocally demonstrated or where a degraded or nonconforming condition results in a judgment that the equipment is operable but there are remaining concerns or uncertainties. Station procedures address the detailed process; however, the station procedures generally agree with the guidance provided in NRC Generic Letter 91-18 and the approach described here.

PROCESS DESCRIPTION

ISSUE SCREENING

When an operability issue is identified, Operations expeditiously performs an issue screening.

Completion of the issue screening will determine if the SSC is:

Operable with no concerns.

Inoperable. Review for reportability.

Operable with potential concerns. This determination will require a Concern Screening to be performed by Engineering.

CONCERN SCREENING

Concern Screenings are performed by knowledgeable qualified engineers to determine whether an operability concern exists. Screenings are performed using detailed guidance.

Completion of the concern screening will determine if the SSC is:

Inoperable.

Operable.

Concern confirmed, perform operability evaluation.

OPERABILITY EVALUATION

Operability evaluations are performed by knowledgeable qualified engineers using detailed guidance.

Completion of the Operability Evaluation will determine:

If compensatory actions are required to maintain functionality.

If corrective actions are required to restore full qualification.

REVIEWS

The Operability evaluation is reviewed by Engineering and Station Management.

CLOSURE

An operability determination is open as long as the degraded or non-conforming condition exists. The operability can only be "closed" when it can be shown that the SSC has been repaired or modified to meet the original full qualification or the design bases has been changed via a modification and/or UFSAR change so that the "as-found" condition now meets full qualification.

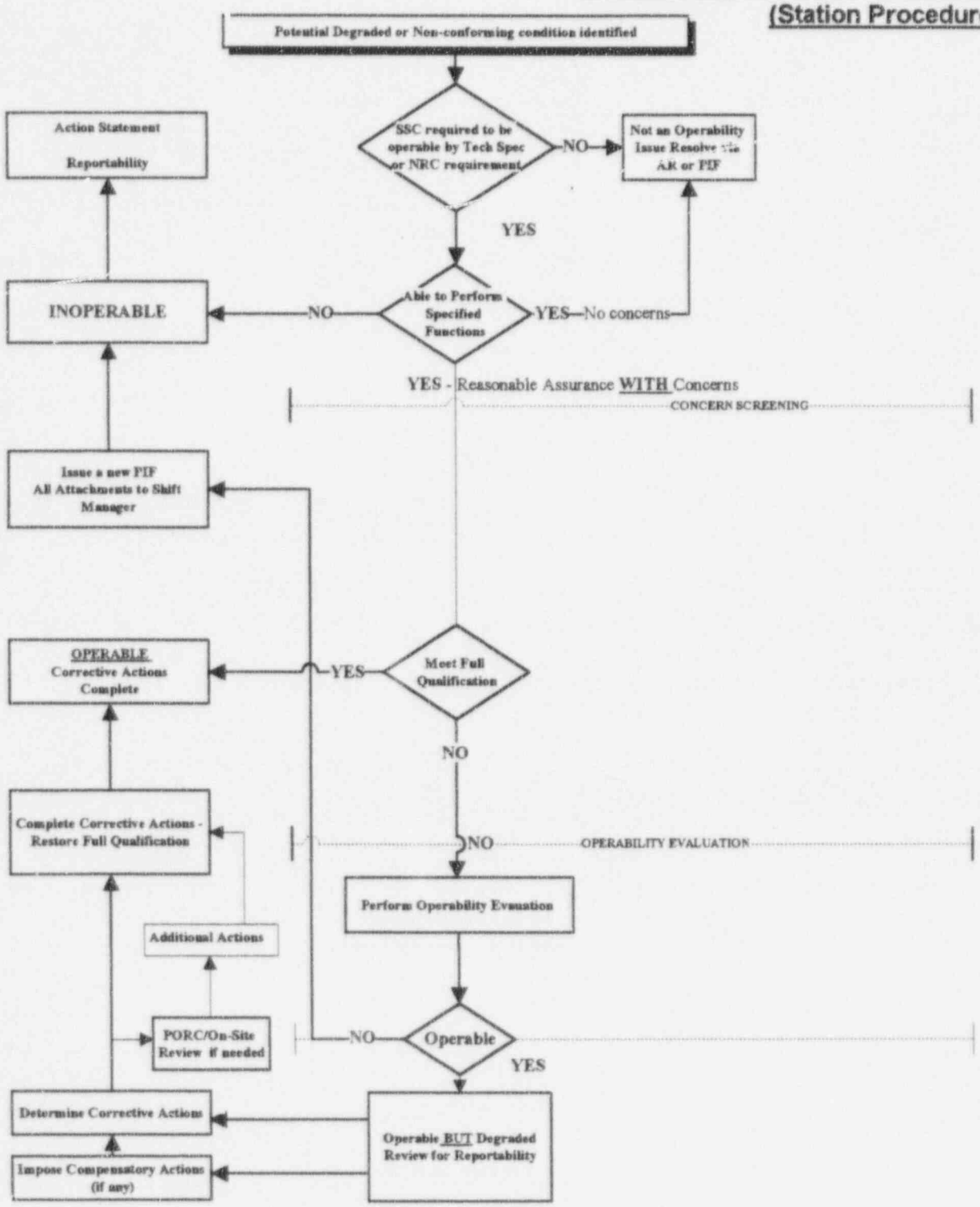
CHECKS AND BALANCES

An Operability Evaluation undergoes an overview review by a specially constituted team with representatives from Operations, Engineering, Regulatory Assurance and Station Management. In some cases, it is subjected to review by the Plant Operations Review Committee (PORC) or On-Site Review Committee.

RECENT/PLANNED IMPROVEMENTS

ComEd is in the process of adopting a uniform Operability Determination Process across all six sites.

Flowchart 18
Operability Determination Process
(Station Procedure)



UFSAR Update Review, Process 19

PURPOSE

Changes made to the facility, equipment, analysis, procedures, programs, or organizations that change the description or drawing in the UFSAR, require a UFSAR Change to be initiated. The impact of UFSAR changes to the station's Licensing Bases is controlled through detailed preparation and review processes as described below:

PROCESS DESCRIPTION

Changes to the UFSAR can result from the design change modification process, or they can be self-generated as part of a general UFSAR update program through UFSAR reviews associated with the normal work process, or regulatory or self-assessments. The impact of UFSAR changes originated as part of the modification process is addressed as part of that process. The process addressed here describes how self-generated UFSAR changes are evaluated and implemented.

CHANGE PREPARATION

The initiator of a potential UFSAR Change thoroughly researches the change to determine whether other sections of the UFSAR are also affected by the change and lists all affected UFSAR sections, tables and figures associated with the change. Proposed changes should also be investigated to determine that they don't conflict with the Technical Specifications, system design bases or other station commitments.

10 CFR 50.59 SAFETY EVALUATION

10 CFR 50.59 Safety Evaluations are performed to determine if the UFSAR Change could involve an Unreviewed Safety Question or a change to the Technical Specifications.

All Technical UFSAR Changes receive a 10 CFR 50.59 Safety Evaluation, unless the change has been approved by the NRC and an SER was issued.

TECHNICAL REVIEW

All UFSAR Changes receive as a minimum a Technical Review to verify that the proposed information is technically correct. Technical Reviews are performed by individuals knowledgeable in the subject matter, and a member of the cognizant department for which the change is intended must sign as one of the Technical Reviewers.

CHECKS AND BALANCES

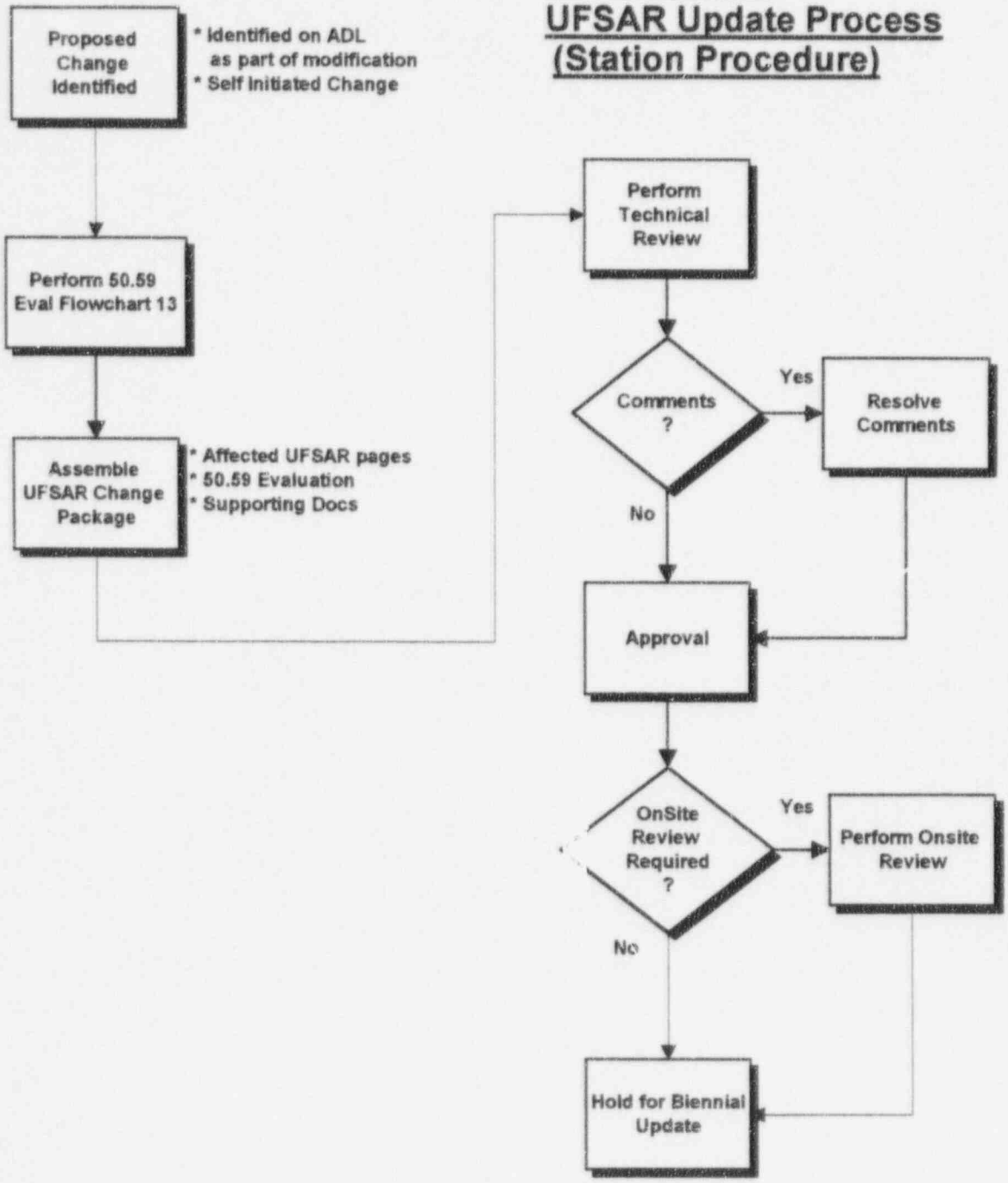
The Technical Review performed at Byron Station for all UFSAR changes provides an important checkpoint in the process to ensure regulatory compliance and to maintain license control. Also,

Byron and Braidwood share a common UFSAR update process because they share a common UFSAR and both sites review and approve all changes.

In addition, all engineering related UFSAR changes are reviewed and approved by a cognizant Engineering Supervisor. This provides an important administrative and technical checkpoint in the process.

The optional On-Site Review also provides an additional checkpoint in the process.

Flowchart 19
UFSAR Update Process
(Station Procedure)



Out-of-Service/Return-to-Service, Process 20

PURPOSE

This process provides an overview of the common approach utilized to initiate and remove an equipment Out-Of-Service. The detailed control procedures are station procedures, and are unique to each station.

PROCESS DESCRIPTION

The following is an outline of the equipment Out-of-Service (OOS) and Return-to-Service (RTS) process. It is controlled via station procedures.

PLACEMENT OF OOS

Station personnel may initiate an OOS Request to perform work safely on station equipment or to otherwise maintain and control abnormal configurations. This process is managed through ComEd's Electronic Work Control System (EWCS).

1. Work Groups requesting the OOS are responsible to sufficiently define the scope of the work to allow the Operations Department to develop an adequate OOS.
2. Qualification requirements are established for individuals who prepare and review OOS. Controlled documents and drawings are used to ensure accuracy of prepared OOS. When controlled drawings are unavailable, the OOS will be walked down in the field to ensure accuracy. A second qualified OOS Preparer independently verifies the OOS as correct.
3. The OOS is reviewed by an SRO licensed operator to identify Technical Specification (Tech Spec), containment related, fire protection/Appendix R and other issues.
4. A SRO licensed Unit Supervisor in the Control Room conducts an independent review and weighs the impact of the OOS on the probabilistic risk assessment for the Unit.
5. A RO licensed Nuclear Station Operator (NSO) reviews and verifies the OOS is correct for the current plant conditions and will brief the Operations personnel positioning equipment and hanging the OOS cards.
6. Both licensed and non-licensed operators may place OOS cards. All cards are hung and then independently verified unless waived by Unit Supervisor per station procedure.
7. The Work Group Supervisor is responsible to verify the OOS has been correctly hung and is adequate for the scope of the work.

RETURN TO SERVICE

When work is completed, a Return-To-Service (RTS) Request initiates removal of the OOS.

1. A qualified OOS preparer reviews controlled documents and drawings to prepare the RTS and determine repositioning requirements for equipment.
2. A second OOS Preparer verifies the RTS is correct.
3. RTS is SRO reviewed to identify potential Tech Spec/Containment issues.
4. RTS is verified by the Unit Supervisor to ensure Tech Spec/Containment issues have been identified and that equipment repositioning requirements are appropriate.
5. A NSO will review the RTS and brief the operators who will reposition equipment and remove the OOS cards.
6. All equipment is repositioned and OOS cards are removed with independently verification unless waived by Unit Supervisor per station procedure.

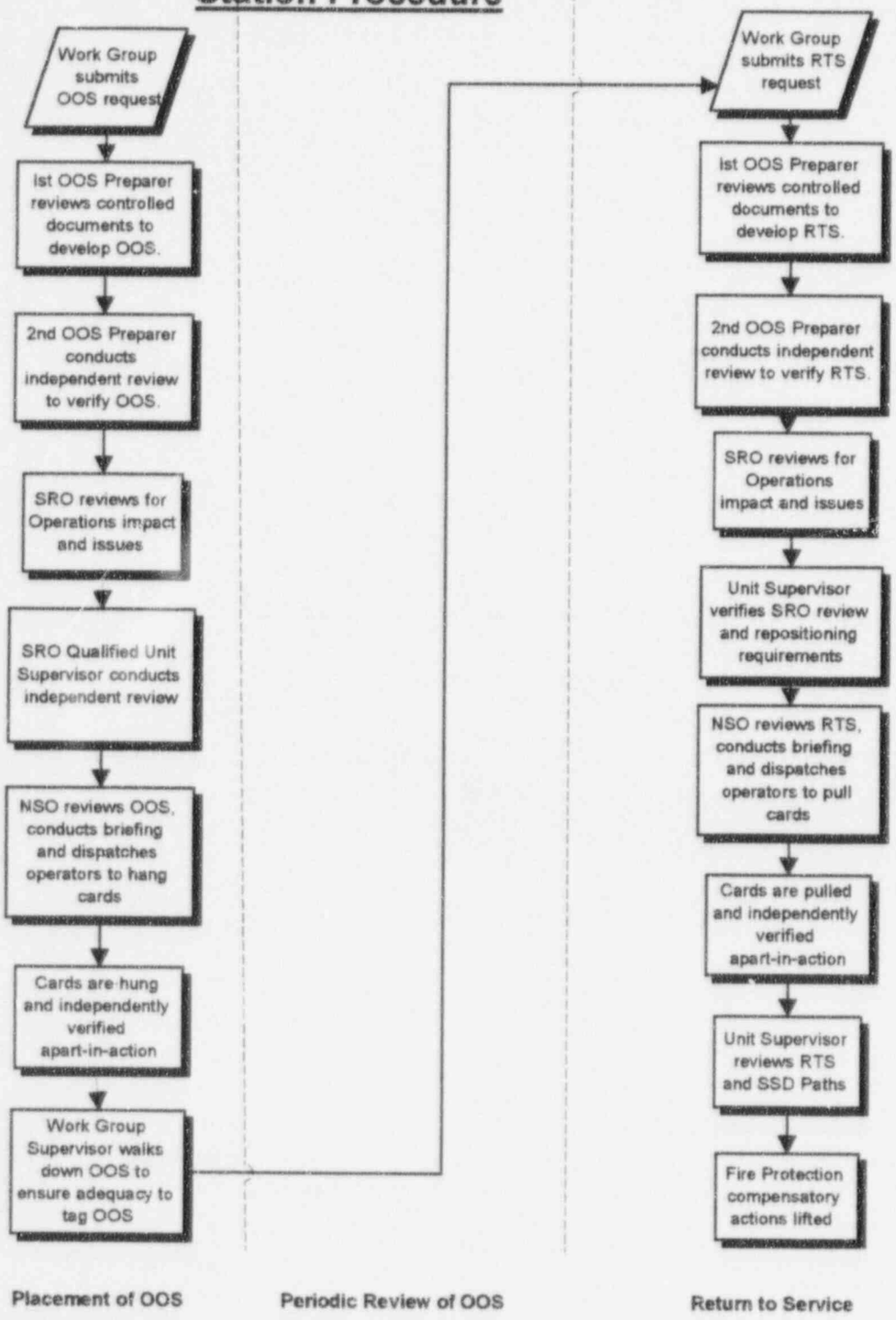
CHECKS AND BALANCES

Independent verification is used throughout the OOS program. There are two OOS Preparers and each is responsible to independently review controlled documents and drawings to satisfy themselves that the points of isolation and special instructions are correct. Technical Specification, containment impacts fire protection/Appendix R and other operation impact and issues are also independently reviewed by SRO licensed operators. When equipment is positioned and cards are hung during OOS or RTS, two operators are normally assigned to perform independent verification. The review by both the Unit Supervisor and NSO considers potential impacts of the OOS or RTS on the current plant configuration. The Work Group Supervisor is responsible to ensure that the OOS is appropriate for the scope of work to ensure protection of the equipment as well as personnel safety. The periodic review of OOS ensures that OOS have received a 10 CFR 50.59 Screening/Evaluation to ensure the level of plant safety is not degraded by the duration of the OOS and equipment is maintained in the correct OOS position.

RECENT/PLANNED IMPROVEMENTS

ComEd has initiated a corporate-wide standardization of the OOS process. The new process is being designed to eliminate many issues common to all sites, and is exploring the use of an all electronic version of the Out-of-Service Program.

Flowchart 20
Out-of-Service/Return-to-Service Process
Station Procedure



Appendix III - Nuclear Fuel Services' Design Processes

The Nuclear Fuel Services (NFS) Department is the major ComEd Corporate Engineering organization providing production services to the ComEd nuclear stations. In the past, its functions were performed by a separate service organization that was not a part of corporate engineering and was under separate management. Consequently, when NFS was merged into the Nuclear Engineering Services Department under the direction of the Engineering Vice President, it already had unique processes and procedures that migrated with it to the new organization. This Appendix addresses those unique NFS processes that impact design bases and configuration control.

In addition, in recent years, NFS has had an increasingly important role in establishing and maintaining the design bases. New reactor fuel designs, new fuel vendors, changes to the core configuration, changes to core components and changes to the refueling cycles can have impacts on the thermal-hydraulic and transient analysis that form the bases of the safety analyses and evaluations. These important roles are discussed in this Appendix.

Organization and Responsibilities:

The NFS Department has lead responsibility for Core Reload Design and other reactor core components for all six nuclear stations. The NFS Chief Nuclear Engineer and the NFS Supervisors plan, direct and monitor all activities related to Core Reload Design. The NFS Chief Nuclear Engineer reports directly to the Engineering Vice President. Reporting to the NFS Chief Nuclear Engineer are Supervisors for the following areas (PWR and BWR): Support Services, Nuclear Design, and Safety Analysis.

The PWR and BWR Support Services Supervisors administer the technical projects involving the fuel, reactor core and core components in support of the Core Reload Design of the reactors. The PWR and BWR Nuclear Design Supervisors administer activities related to reactor neutronic analyses which are required for the Core Reload Design. The PWR and BWR Safety Analysis Supervisors administer the activities related to thermal-hydraulic and transient analysis for the reload safety evaluations of each of the operating nuclear reactors.

A Reload Licensing Engineer (RLE) provides oversight and input as needed for the licensing aspects of the reload process. A Fuel Reliability Engineer (FRE) provides oversight and input as needed in the area of fuel reliability. A FRE monitors fuel performance and provides recommendations to the stations on activities such as fuel inspections and reconstitution. A FRE also reviews significant changes to fuel designs and manufacturing processes prior to their implementation. Both, the RLE(s) and FRE(s) report directly to the Chief Nuclear Engineer.

The Site Vice President and Senior Station Management are responsible for providing oversight review and concurrence with the reactor core design. This includes significant changes in unit operation philosophy (such as 24 month cycles) and fuel design changes. Additionally, they

supply corporate and station goals to be used in the design of the reload (such as the cycle startup/shutdown dates and anticipated operating capacity factor).

The Station Reactor Engineer administers the on-site Core Reload Design activities related to design input, fuel and component handling, core loading, startup testing and operations support. The Reactor Engineer takes functional direction from the NFS Chief Nuclear Engineer in matters related to Core Reload Design. The Site Engineering Manager is responsible for engineering activities at the station. Site Engineering provides input to the Core Reload Design process by identifying any plant modifications or changes which may affect the Core Reload Design.

Onsite Review is responsible for performing a review of the Core Reload Design 50.59 package and/or any license amendments produced in the Core Reload Design process. Offsite Review is responsible for fulfilling the Offsite Review and Investigative Function, including the review of changes to procedures, equipment or systems as described in the Safety Analysis Report. Offsite Review is responsible for performing a review of the Core Reload Design 50.59 package and/or any license amendments produced in the Core Reload Design process.

The Fuel Vendors are responsible for the mechanical design and fabrication of the fuel assemblies, LOCA Analysis of record and maintenance of the Core Reload Design capabilities required by the Fuel Contract and Vendor Interaction Procedures or Guidelines. Fuel Vendors must maintain approved Quality Assurance programs for their design work, which may include some or all of the nuclear design and safety analysis scope if requested.

Core Reload Design Process (Process 1):

Note: For the purposes of this discussion, the term "Fuel Vendor" is applied to the organization responsible for the fabrication of the fuel and delegated to perform the required core design and licensing analyses. ComEd currently performs the core design and is in the process of licensing the capability for performing the cycle specific transient analyses.

The planned completion date of the NFS Reload Design Safety Evaluation (including UFSAR changes and COLR) is dependent upon whether or not a change to the Technical Specifications is required and, if so, its complexity. Requests for Technical Specification Amendments are made as early as practical with the objective of providing sufficient lead time for NRC review and approval.

Normally, the preliminary core design, including fuel bundle design, the goals for the operating cycle performance and the Reload Licensing Schedule are reviewed with Senior Station Management. This review permits Senior Station Management to participate in the review and approval of the reactor core design including significant changes in unit operation philosophy (such as 24 month cycles) and fuel design and/or core component changes. Note that this review meeting is in the process of being enhanced as a result of recommendations from a recent industry (INPO) managers conference.

The Station Reactor Engineer, NFS Support Services and Safety Analysis Cognizant Engineers coordinate and review the transient analysis parameters and LOCA analysis parameters.

The Reload Design Initialization (RDI) process sets the scope and ground rules for the reload design. The RDI process is broken into two parts:

a) The RDI process identifies plant changes such as modifications, Technical Specification amendments and setpoint changes which could potentially affect the design or schedule. The RDI also identifies any fuel design changes or first-of-a-kind applications.

b) The RDI process also determines how the proposed reload design would affect the plant. The RDI process identifies any supporting activities which must occur to support the reload design. Supporting activities include setpoint changes, license amendments, training, procedure changes, special tests and others. The RDI process tracks to completion or resolution each of these changes.

The assumptions and conditions identified in the RDI process are applied in the Core Reload Design process. The Reload Design Safety Evaluation (10 CFR 50.59 for the reload design) confirms that these inputs do not create an unreviewed safety question. The assumptions and conditions are again reviewed prior to criticality in the Reload Design Finalization (RDF) process (discussed below) and incorporated into the Reload Design Safety Evaluation.

When the draft licensing documents are received from the "Fuel Vendor," the Station Reactor Engineer and the Support, Safety Analysis and Nuclear Design Cognizant Engineers perform a detailed review of the draft reload licensing documents. The first action taken when reviewing the results of the licensing analyses is to evaluate the trends by comparing the results to previous reload analyses.

NFS completes a separate evaluation for any new fuel or core component designs under the Nuclear Fuel and Component Design and Fabrication Control Process (see below). This evaluation typically is referenced by the NFS Reload Design Safety Evaluation.

The Nuclear Design Engineer verifies that the final Fuel Assembly Design Package and Nuclear Design Report properly reflects the fuel assembly neutronic designs established for the reload.

Once the reload licensing documents are finalized, they are transmitted to the station as a Nuclear Design Information Transmittal (NDIT).

The Cognizant Support Engineer, with the support of the other review team members, develops the NFS Reload Design Safety Evaluation, including related documents such as UFSAR page mark-ups. The objective of the Safety Evaluation is to review and document the essential aspects of the reload, including fuel design or component changes, with sufficient detail to ensure no unreviewed safety questions exist in accordance with 10 CFR 50.59. An Independent Review by

another qualified Engineer of this package is conducted in accordance with the Controlled Work process (see below).

The Reload Design Finalization (RDF) process is performed to confirm that the assumptions used for the design, analysis, and supporting activities are still appropriate considering the actual conditions and that the required supporting activities (identified during the RDI) are completed or will be completed as required.

A Station Onsite Review and Offsite Review are conducted on the Core Reload Design 50.59 package.

Upon completion of the core loading, the core configuration is verified by the performance of an as-loaded fuel assembly serial number surveillance. Typically, an underwater camera is used and the results are video taped. The Reload Licensing Loading Pattern, used for all licensing evaluations, is the acceptance criteria bases for this review. This surveillance is witnessed by a member of the NFS staff using an independently obtained copy of the Reload Licensing Loading Pattern.

During the latter stages of the refuel outage, the station performs an Onsite Review of the outage activities. A subsection of this review is a verification that the assumptions used for the design, analysis, and supporting activities are still appropriate considering the actual conditions and that the required supporting activities (identified during the RDI) are completed or will be completed as required.

Upon completion of the refuel outage, unit startup commences. Various startup tests are performed in accordance with the station's Technical Specifications or other administrative controls. Additionally, tests are performed as required by the Core Reload Design process. The results of these tests are evaluated to provide assurances that the design is valid by comparing test results to design values for key parameters.

Nuclear Fuel and Component Design and Fabrication Control Process (Process 2):

The Fuel and Component Design and Fabrication Control Process involves the technical review of all significant changes to the design of the fuel assembly. This design review covers, as a minimum, the potential impact of the change on plant safety and transients, interfaces, reliability, and performance. A Fuel Reliability Engineer (FRE) has the primary responsibility for implementation of this process. Other areas of NFS have the responsibility to provide personnel to assist in or lead the review of nuclear fuel or core component design changes as agreed upon between the NFS Chief Nuclear Engineer, NFS Supervisors, and a FRE.

Uranium enrichment and burnable absorber content vary from cycle to cycle to accommodate cycle energy requirements. These parameters are specified by Nuclear Design and may be included under this process if their values are outside previously utilized ranges and there is a possible affect on safety or transient analysis, fuel rod performance, etc.

The significance of the change is determined by a FRE or designee by reviewing the drawing or specification changes provided by the vendor. Any questions or comments about the design changes should be discussed with vendor personnel.

For Significant Design Changes, a more rigorous review process is required, as follows:

A Design Review Team is formed consisting of NFS personnel, appropriate station personnel and, when needed, appropriate technical experts from outside NFS. Documentation of the review is maintained including any notes or minutes from meetings and telecommunications with vendor personnel or expert consultants on the design change.

The Design Review Team thoroughly reviews the design change and all documentation provided by the vendor to support the change. In addition, the Design Review Team requests any additional information from the vendor which it believes would assist in the review. Information such as design analyses, design bases, prototype testing, Lead Test Assembly (LTA) experience, the vendor's qualification of the design change and fuel fabrication process changes associated with the design change are typically requested to assist in the evaluation.

The following conditions are those that typically require NRC approval prior to implementation of a fuel or component design change:

- Any hardware change that results in a design that is different than that described in the Technical Specifications (e.g. different clad material, fuel or absorber material).

- Any design change that results in an unreviewed safety question per the criteria of 10 CFR 50.59.

- Any hardware change that is not bounded by an applicable ComEd or Vendor topical report (e.g. a spacer grid design change that requires a new CHF/CPR correlation).

After resolution of all technical issues related to the design change, the Design Review Team determines if the design change is technically acceptable for application at ComEd plants. In some cases the Design Review Team will also determine if the design change is financially attractive to ComEd (i.e. there is a justified economic payback if the change involves a cost increase to the price of the fuel).

If the design change is acceptable to the Design Review Team, station concurrence with the change is obtained. Significant design changes are reviewed and approved by Senior Station Management.

The Design Review Team prepares a report of their review of the design change. This report details all the technical issues associated with the design change and their resolution. The report is typically signed by all team members. The Design Review Report is considered Controlled Work.

The Design Review Team Leader prepares a memo to the ComEd Buyer for the NFS Chief Nuclear Engineer's signature which accepts or rejects the design change. The memo lists any limitations or conditions which the team believes are needed to make the design change acceptable for use in ComEd plants or contains the reasons for rejection of the design change, if necessary.

The FRE follows up to assure that all limitations and conditions agreed to between the vendor and the Design Review Team are followed both in the designing and manufacturing as well as the handling and use of the fuel or component at the plant.

Nuclear Fuel Services Controlled Work Process (Process 3):

Controlled Work is a calculation or analysis, or formal evaluation, review, response or recommendation, or change thereto, which is:

- Important to safety in the design or operation of a fuel rod, fuel assembly, or reactor core, or in the design or operation of a plant system, subsystem or component; or,
- Used to generate information which will be sent to the NRC in support of ComEd submittals; or,
- Used to support an NFS, Station or other ComEd department Safety Evaluation, Significant Hazards Evaluation, Technical Specification or FSAR change or interpretation thereof; or,
- Used in the generation of Special Nuclear Material accountability information.

All Controlled Work receives an Independent Review by a qualified Engineer.

A Controlled Analysis is any NFS calculation that meets one or more criteria of Controlled Work.

A Routine Controlled Analysis is a Controlled Analysis which is performed according to a procedure for a recurring application.

A Special Controlled Analysis is a Controlled Analysis for which no procedure has been written, or for which a procedure cannot be followed without alteration that affects the intent of the procedure or the margin of safety.

A Routine External Analysis is a standard, recurring analysis performed external to ComEd which meets one or more criteria of Controlled Work and which has been performed in accordance with the external organization's ComEd-approved Quality Assurance program.

A Special External Analysis is a non-routine, infrequently performed, or first of a kind analysis performed external to ComEd which meets one or more of the criteria for Controlled Work.

An Additional Review (AR) is required for all Special External Analyses, after completion of the initial Acceptance Review. For the other types of Controlled Work, the NFS Supervisor shall determine whether an Additional Review (AR) and/or a Special Review Team (SRT) is warranted and shall document this conclusion. Examples of Controlled Work that may require review by a SRT are:

- First-of-a-kind application of a substantially new methodology or design.
- First application of a Special Controlled Analysis or Special External Analysis that is particularly significant, or that has a direct and significant impact on a Technical Specification or that is required for NRC submittal.
- Special Analyses or safety reviews or recommendations that would result in a major change in station operation, Special Nuclear Material accountability, or reactivity management.

Review of Problem Identification Forms (PIFs)

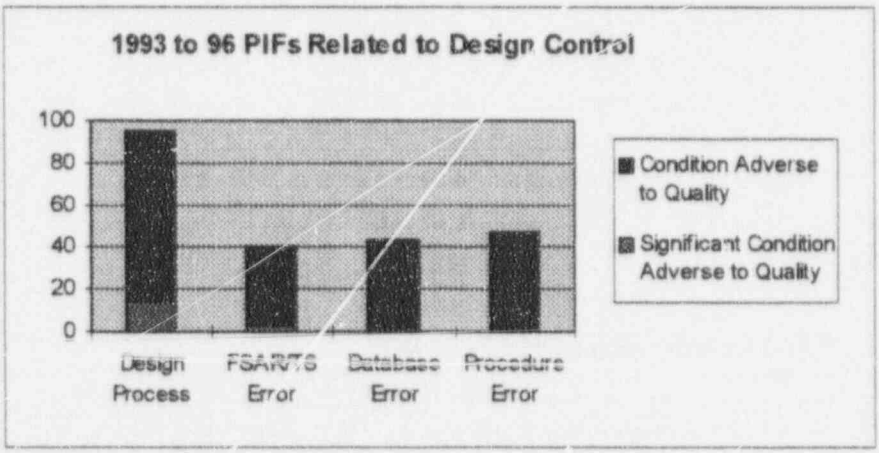
A review was performed of NFS generated PIFs from 1993 (the first year the PIF process was used in NFS) to present (November 8, 1996). As described in Action (d), the PIF process is common to all six nuclear stations and is also used by NFS to identify, document, assess, and correct design bases and other nonconformances. Nearly 50% of the NFS generated PIFs were associated with the reload design process (RDP). A review of each year's PIF log demonstrated that this trend is also prevalent on a yearly basis. Over the three and a half year period, nearly half of the design bases deficiencies were equally distributed in the areas of the licensing bases documents (UFSAR and Technical Specifications), databases (typically computer data files) and procedures. The remaining 50% are associated with the design bases process itself. Approximately 10% of the reload design process PIFs were categorized as significant and received a heightened level of investigation. It should be noted that RDP PIFs that had the potential to result in a reportable issue per 10 CFR 50.72 or 73 were typically issued by the affected station independent of the location of the identifying organization.

The RDP PIFs covered a spectrum of issues; from minor errors caught during the Independent Review process to significant process deficiencies that resulted in notable process enhancements. The age of the deficiencies also ranged widely; from inaccuracies in currently open evaluations to original licensing bases analyses.

Significant design bases process enhancements that resulted from RDP PIF investigations include:

- Created a transient input parameter list.

- Created a reload design initialization/control procedure.
- Developed reload interaction agreement with Fuel Vendor for pertinent fuel rod design information.
- Upgraded procedure for Controlled Work to improve required handling and review of all external documents including those classified as routine design.
- Changed the threshold for writing PIFs to require that any anomalies identified consistent with a "controlled work" review be PIF'd.
- Developed a Quality Software Control Process. The various stages of testing, validation, operation, maintenance and upgrades were defined and a list of approved quality software developed, communicated and maintained.



Summary of Major Audit Findings and Corrective Action

Nuclear Fuel Services (NFS) and the Nuclear Engineering Groups at the stations, as the owners of the Reload Design Process, participate in an aggressive design control audit and technical review program. NFS and the Nuclear Engineering Groups participate in audits of the ComEd nuclear stations, fuel and core component vendors and licensing analyses Architect Engineers (A/Es). For ComEd internal audits, the Site Quality Verification (SQV) department is typically the coordinating organization. For external audits, the Supplier Evaluation Services (SES) department is typically the coordinating organization. Some of the external audits are conducted as a joint audit by a collection of utilities. All audits are undertaken periodically or as a special review as the result of an adverse trend.

Typically, members of NFS and/or the Station Nuclear Engineering Groups participate in internal and external audits as the audit team's Technical Expert(s). ComEd internal audits have included reviews of the reload design process and the Reactivity Management program. External audits have included issues from fuel and nuclear component fabrication (at the manufacturing facility) to licensing analyses. Findings and Recommendations are identified and conveyed to the auditee. Some of the more significant findings (Level II) are listed as follows:

- Using an unapproved procedure to make changes to controlled documents without making a revision change to the document.
- Reference files used during testing of a revision to the Core Monitoring Software were not completely reviewed.
- The calculation notebook to support the application of Traversing Incore Probe (TIP) machine data substitution methodology was not completed.

As part of the transition to Siemens Power Corporation (SPC) ATRIUM-9B fuel at ComEd's BWRs, increased vendor special audits and technical reviews have been and are continuing to take place at SPC's offices/facilities due to the introduction of the new fuel type and licensing methodologies. Examples of these include a technical review of the LaSalle Equipment Out Of Service Analysis and a technical review of the Quad Cities LOCA/ECCS analysis.

The Reload Design Process has also received both internally and externally originated audits. These audits are initiated both periodically as well as when a trend is identified. Over the last few years, the Reload Design Process has been the subject of numerous internal and INPO audits as well as two NRC inspections. Overall, the Reload Design Process has been found by the NRC to be satisfactory. The 1992 inspection[1] found a strength in.

"Communications between the station personnel (PWR) and NFS was a strength and included:

- The weekly conference call with the three Lead Nuclear Engineers from the three PWR stations.
- A single NFS contact for each station contributed to effective and efficient communications.
- Direct access (using the paging system and home telephone numbers) and availability of Technical Staff (NFS) personnel during off-normal hours and weekends."

The 1994 inspection[2] also found the Reload Design Process to be satisfactory:

> "Overall, we found that the conduct of activities related to the development of core reload analysis for the ComEd stations were good. The Corporate Nuclear Fuel Services department was found to be a technically strong, interactive organization, providing good communications and support to the nuclear engineering groups at each of ComEd's nuclear power plants. We were encouraged by the depth and extent of the root cause investigation and corrective actions taken in response to the June, 1994 failure to install hafnium rod inserts event."

However, weaknesses were also identified such as:

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[1] Inspection Reports No. 50-295 / 92012 (DRS); 50-304 / 92012 (DRS); 50-454 / 92010 (DRS); 50-455 / 92010 (DRS); 50-456 / 92010 (DRS); 50-457 / 92010 (DRS), April 27 through May 8, 1992, Routine Inspection of nuclear engineering related activities at both the three PWR plants and at the Nuclear Fuel Services Department.

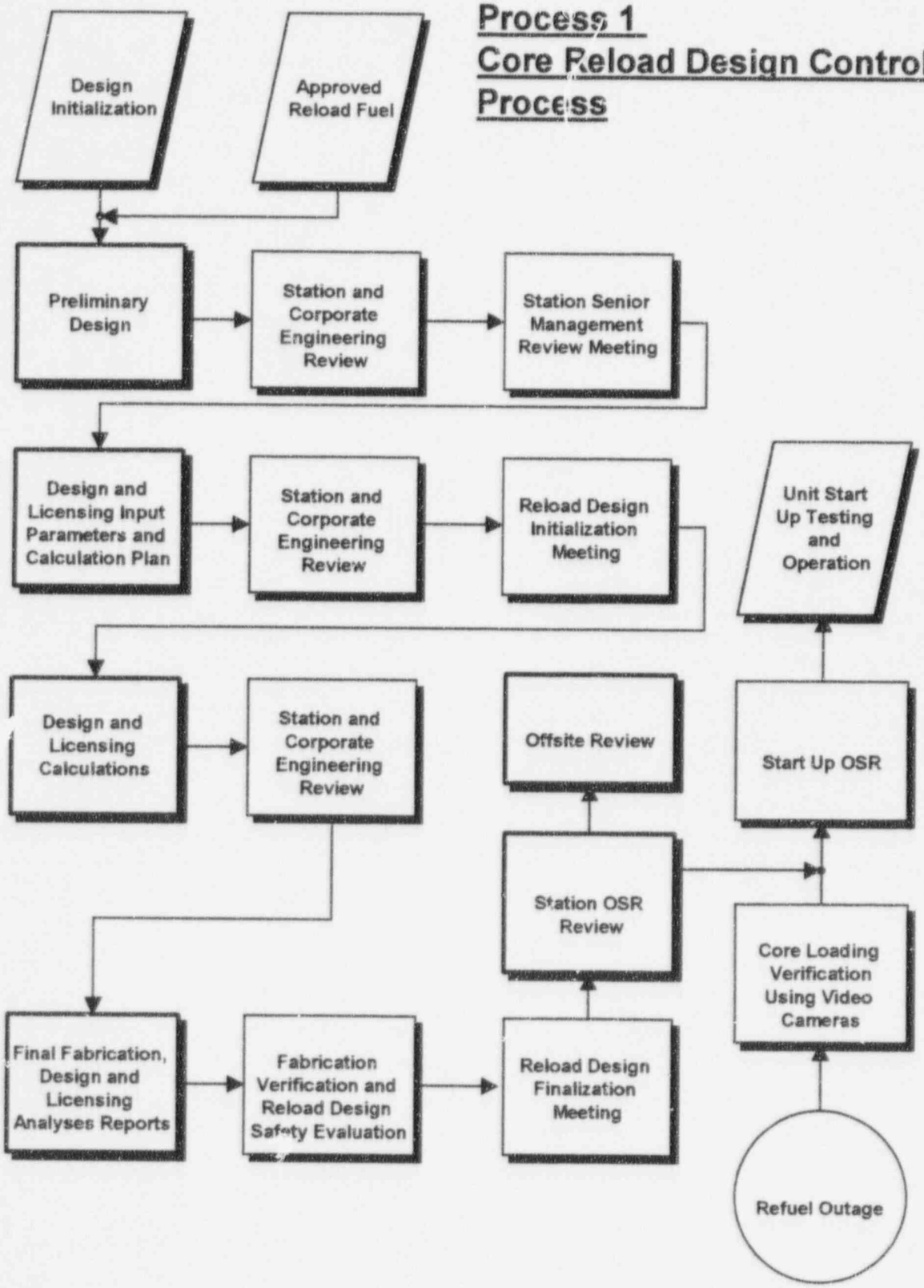
[2] Inspection Reports No. 50-295 / 94022 (DRS); 50-304 / 94022 (DRS), October 17 through October 21, 1994, "Special Inspection of the failure to include Hafnium rod inserts at the Zion Nuclear Power Station and a review of ComEd's Nuclear Fuel Services Organization".

"Most communication for special circumstances and unique issues appear to be verbal"; "Training and qualification was identified as a contributing cause to the reactivity control problem"; and,
"... deficiencies were identified in the areas of Qualified Nuclear Engineer (QNE) training and self-assessment. The QNE training deficiencies involved a lack of clear ownership of the QNE requirements. Additionally, the self-assessment process was of limited benefit to the NFS organization, primarily because this effort was still in the initial stages of development."

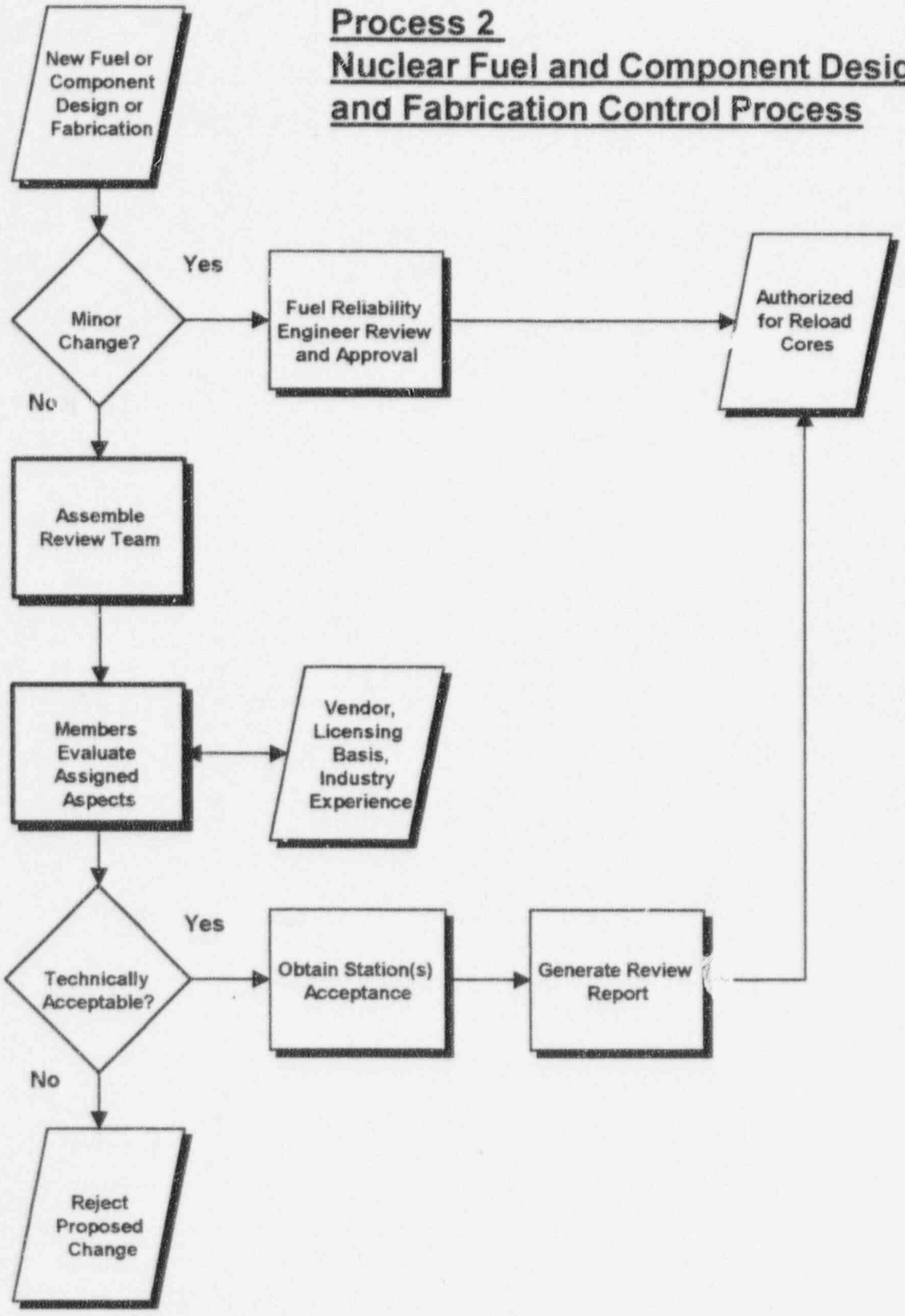
These weaknesses have been and are continuing to be addressed through enhancements to the reload design process.

In addition to corrective actions and process improvements undertaken in response to audits and regulatory findings, NFS is planning to implement a proactive process improvement that was identified from recommendations made at an industry managers conference. A review meeting with Senior Station Management is being added to the Core Reload Design Process. This review meeting provides Senior Management oversight review and approval of the core reload design including significant changes in unit operation philosophy and fuel design changes.

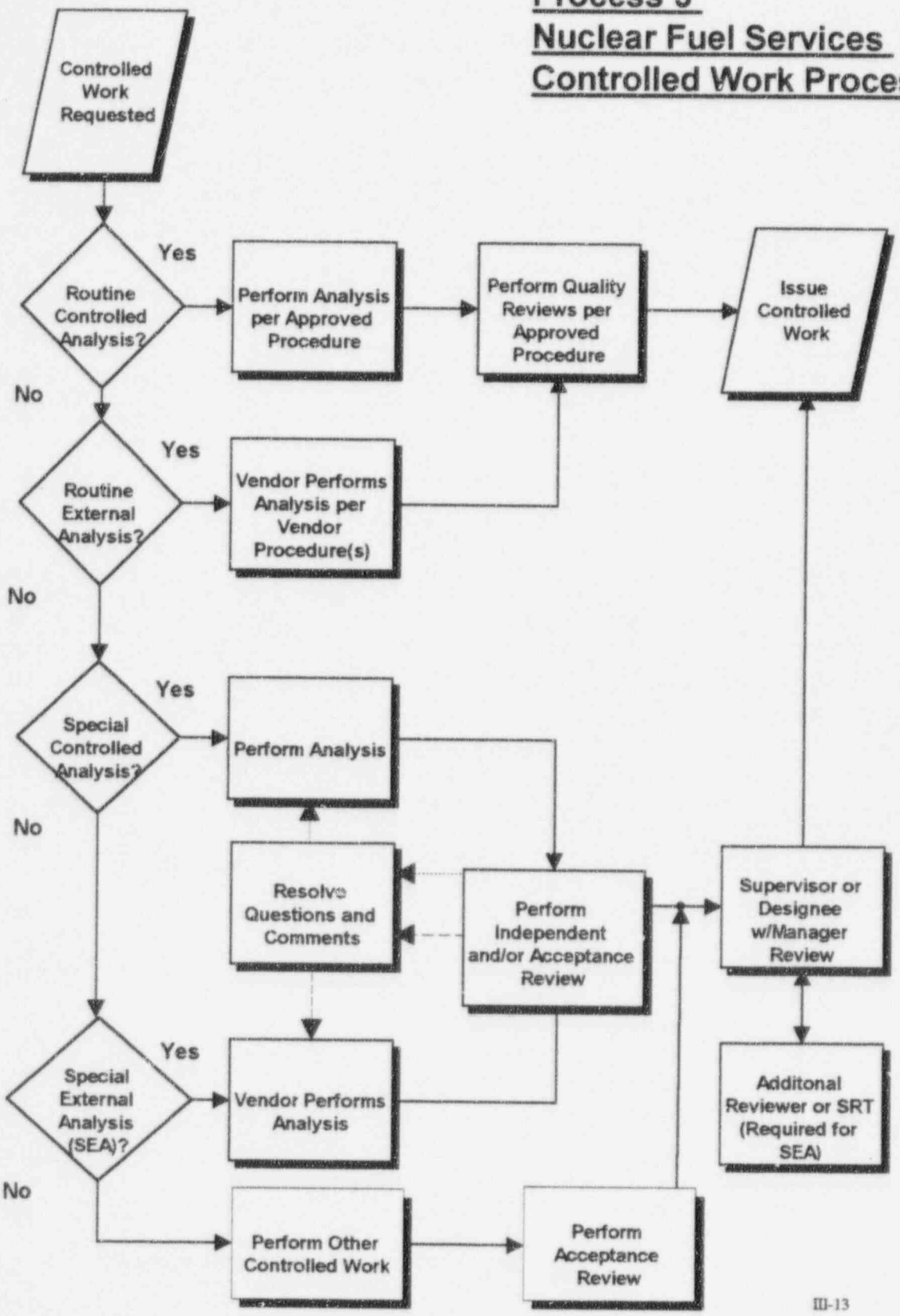
Process 1
Core Reload Design Control Process



Process 2
Nuclear Fuel and Component Design and Fabrication Control Process



Process 3
Nuclear Fuel Services
Controlled Work Process



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