Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701

ComEd

February 6, 1997

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

Subject: Braidwood Station Units 1 and 2
Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Basis Information
NRC Docket Numbers: 50-456 and 50-457

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 Braidwood 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, Braidwood concludes that there is reasonable assurance that its procedures accurately reflect its design bases and that it is configured and operated in a manner which is consistent with its design bases, as defined in 10 CFR 50.2, or as otherwise permitted under the NRC's regulations.

Braidwood'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. Validation and review of the response was conducted at several levels, including reviews by Braidwood site management, a ComEd Corporate Team, and Braidwood's Plant Operating Review Committee (PORC). Finally, an external review team comprised of high-level

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individuals who have extensive experience with the nuclear regulatory process and are not involved in ComEd's day-to-day activities provided an independent assessment of the response. These processes for preparing and reviewing the response provide added 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 Braidwood 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, "Design Control and 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

This response is intended to provide detailed information on the above listed action items. This response captures and condenses a substantial body of information and 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 with regard to ComEd's commitment to future actions regarding 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

Braidwood's conclusion that there is reasonable assurance that it is configured and operated consistent with its design bases is based on several factors First, the plant was determined to meet its design bases when it was licensed. ComEd certified that Braidwood 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. Braidwood received the benefit of the procedures which had been developed and used successfully in the first 18 months of Byron operation The product of the combined inspection and assessment efforts on Byron and Braidwood provided a high level of assurance that Braidwood'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 which 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. ComEd 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. Braidwood also has responded as expected during unplanned events and transients.

Self-assessments, ComEd Quality Assurance audits, NRC inspections, and third party reviews have examined not only the processes implemented to maintain procedures and SSC's consistent with the design bases but also the procedures and the SSC's themselves. For the most part, these reviews have corroborated that the plant's procedures accurately translate design bases information and that its structures, systems, and components are consistent with their design bases. Where substantial discrepancies have been identified, their root causes and extents of occurrence have been determined, the discrepancies have been corrected, and the procedures which permitted them to occur have been strengthened. This experience leads Braidwood to conclude that its configuration management processes, as supported by its programs for operational monitoring and walkdowns, inspections 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 had been designed and constructed consistent with the design bases information contained in the UFSAR 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, the design engineering organizations currently in place at Byron and Braidwood contain many personnel who participated directly in the original design and construction of the plants. 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 or reconstitution programs. However, Byron and Braidwood have undertaken specific initiatives which have reconstituted or established more specific calculations which implement the design bases where necessary, such as in the development of motor operated valve (MOV) program.

Future Action

Despite the substantial bases for concluding that Braidwood is configured and operated consistent with its design bases, recent inspections have heightened ComEd's concerns in this area. 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 to Mr A. Bill Beach (NRC - Region III) on November 12, 1996, (Reference (b)) that all six ComEd Nuclear Stations would implement certain actions which would enhance ComEd's assurance that the nuclear stations are operated and maintained consistent with their design bases These actions include reviews of the top ten risk significant systems, establishing an onsite Engineering Assurance Group, expanding our Quality Assurance audits of our engineering contractors, and validating or reconstituting critical calculations when necessary to support ongoing operations.

In addition to the actions described in Reference (b), ComEd will undertake other actions to further improve the quality and consistency of plant procedures and physical configuration with design bases For Braidwood, these actions will include enhancing the indexing of the existing design information to facilitate its use, and review of the UFSAR for consistency with plant procedures and configuration (continuation of the reviews for selected systems which already have been performed) The effectiveness of these programs, as well as Braidwood's overall design control process, will be evaluated through ongoing, SSFI-type inspections

In conclusion, ComEd is dedicated to the safe operation of its nuclear power plants We clearly recognize the importance of operating and maintaining Braidwood Station in conformance with the design bases The commitments in the Reference (b) and (c) letters, and the ongoing oversight by Site and Corporate Quality Verification and Engineering Assurance 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

H. Gene Stanley
Site Vice President
Braidwood Generating Station

TJM/tb

Attachment

cc: A. B Beach, Regional Administrator - RIII
J. Callan, Executive Director for Operations
S. Collins, Director - NRR
G. Dick, Braidwood Project Manager - NRR
C Phillips, Senior Resident Inspector Braidwood
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 5-9-98

"OFFICIAL SEAL"
MICHELLE KOWALKOWSKI
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 5/9/98

Michelle Kowalkowski
Notary Public

EXECUTIVE SUMMARY

The following provides a brief summary of Braidwood Station's response to the NRC's October 9, 1996 request for information pursuant to 10 CFR 50.54(f) regarding adequacy and availability of design bases information:

Action (a): We have reviewed the Braidwood Station and related corporate engineering design and configuration control processes, including those which implement 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR Part 50. The processes are described in detail in the response in Action (a), as well as in Appendix II. The results of this review support a finding that the scope and extent of these processes are adequate to maintain the plant configuration and performance 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. 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.

Action (c): Our assessment also addressed the rationale for concluding that systems, structure, and component (SSCs) configuration and performance are consistent with the design bases. Based on the formal certification of conformance which was part of the original licensing process; the preoperational verification and testing activities; the ongoing verifications 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 inspections, including the resolution of identified problems and the implementation of improvement initiatives; there is reasonable assurance that Braidwood's SSCs configuration and performance are consistent with the design bases.

Action (d): We have reviewed processes for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence, and reporting to NRC. The processes are described in detail in this response. The results of our review support a finding that the scope and extent of these processes is adequate to provide reasonable assurance that design bases nonconformances are identified and resolved in a timely manner.

Action (e): We further have conducted an assessment of the overall effectiveness of current processes and programs in concluding that the configuration and performance of Braidwood are consistent with the design bases. 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. The bases relied upon in reaching this conclusion involve the following elements: (1) consistency with design bases at the time of licensing; (2) controls in programs and processes which 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) ongoing verifications of consistency with design bases as part of normal plant activities; (5) verification of consistency with design bases through self-assessments, NRC inspections and third-party reviews; (6) processes for identifying discrepancies and implementing corrective actions, and (7) continuation of activities that assure ongoing consistency with design bases.

Future actions to be taken regarding adequacy and availability of design bases information, design control, configuration management and related areas, will be described under separate cover to the NRC

Braidwood Response to 10 CFR 50.54(f)

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Braidwood Response to 10 CFR 50.54(f)

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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. These processes implement ComEd's configuration management model. It is followed at both the corporate office and the sites.

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 a supervisor in the Engineering Department.

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 that actually implement the processes which constitute the configuration management model. The role of each is summarized below.

NEPs and NSWPs provide corporate expectations for configuration control processes. Their major elements relevant to engineering design and configuration control are summarized in Appendix II. A matrix in that appendix illustrates how the various processes relate to the configuration 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 corporately sponsored for all stations and is discussed in Appendix III.

Station Administrative Procedures provide the site specific 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. As with all procedures, procedural adherence is a clearly communicated management expectation and processes are in place to reinforce that adherence.

Each station's 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 basis requirements through controlled processes for review and approval of the design, its installation, its testing and its operation.

- Assure compliance with 10 CFR 50.59.

- Assure implementation of the FSAR update requirements in 10 CFR 50.71(e).

- Assure that QC inspections and post-modification tests are conducted for modifications.

- Assure the timely 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 basis information and personnel who are familiar with the design bases.

- Assure that personnel are trained.

1.2 Requirements of Procedures Which Control Design Bases

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 in a timely manner 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 a 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 processes conform to Criterion III of 10 CFR 50 Appendix B and include the following provisions:

- The procedures apply to new design work and design changes for safety-related and certain non-safety-related structures, systems, and components.
- 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.
- Design changes are reflected in controlled sets of analyses, specifications, and drawings.

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.
- 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 modifications, operator workarounds, operation with nonconforming conditions) 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 basis and other design documents.
- Changes in the plant are required to be reviewed under Section 50.59 to determine whether an unreviewed safety question exists.
- Responsible personnel are required to receive training in the above procedure(s).

1.2.3 Procedures for implementing Section 50.59 include the following provisions:

- The scope of procedure includes the following: changes to the UFSAR and Technical Specifications, changes to design and operation, procedure changes, temporary modifications, and prolonged operation with degraded and nonconforming conditions.
- Changes are required to be screened to determine whether they involve a change in the UFSAR or Technical Specifications.
- Changes in the UFSAR are required to have a documented 50.59 safety evaluation.
- Safety evaluations are subject to review and approval by supervisors, and on a selected basis, the Plant Operation Review Committee.
- Unreviewed safety questions and changes to the Technical Specifications are required to be submitted to the NRC for approval as part of a license amendment application.
- Responsible personnel are required to receive training in the above procedure.

1.2.4 Procedures for Implementing Section 50.71(e) include the following provisions:

- 50.59 changes are required to be reviewed to determine whether the UFSAR requires updating (e.g., for permanent changes).
- 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 analyses identified above.
- 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.

1.3 Overview of Processes Which Control Design Bases

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

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

The Flowchart attached to this Action (a) response summarizes the relationships of these groups and how they respond to Action (a) requests for 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. For maintenance work, the Action Request (AR) is generally used. For minor, out of pocket (or tool pouch) maintenance activities which do not affect plant design configuration, station procedures control these activities within strict boundaries. 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. PIFs are reviewed by the shift manager or a licensed SRO for any impacts on operability and Technical Specifications. The licensed SRO 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. ARs are reviewed by an experienced, multidisciplinary screening committee, including representatives from Operations, Engineering, and Maintenance, who assign a work priority consistent with the safety significance of the request and any 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 Action (b), parts replacement (Section 1.10) or setpoint change (Section 1.11) processes are used.

Part of the work analyst's responsibility is to perform a verification walkdown prior to starting the work package. This walkdown provides the opportunity to identify any discrepancies between the plant configuration and the design documents, which will be brought to the attention of the engineering department. In this case, the appropriate request form is used to resolve an document/plant discrepancies.

Replacement parts are evaluated by materials engineering personnel under a like-for-like replacement process in accordance with approved procedures. If the work analyst identifies that a like-for-like is not available, he will initiate an ER for a non like-for-like replacement.

Consistency between any 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 pending to correct a potential or actual condition adverse to quality, pending the completion of permanent corrective action. In such cases, operability evaluations (Section 1.15) are performed to assess whether a Structure, System, or Component (SSC) is capable of performing its specified function in its present condition and what, if any, compensatory action is required. Safety evaluations (Section 1.8) may be required depending on safety significance and time required to take the permanent corrective action. Any work activity affecting the power block portion of the station, other than a routine maintenance activity that uses like-for-like spare parts, undergoes a safety evaluation review per 10 CFR 50.59 (Section 1.8).

Temporary Alterations (Section 1.6) 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 assured therefore 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 design change are executed with a work package prepared by the Work Control Group.

The Out of Service (Section 1.12) process ensures that operational plant configuration is controlled consistent with the design bases during performance of maintenance activities. Work is controlled by an existing 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 testing ensures that the work is done properly and that the equipment conforms to applicable design requirements and can be returned to service. Consistency between operation and the plant's design bases is maintained after work is completed by the process used to return equipment to service. If a special test is required it is prepared using the special test procedure.

Technical Specification surveillances, which verify design requirements, are also controlled through the work control process. This work is performed utilizing controlled procedures. Any discrepancies are reported on a Problem Identification Form (PIF), which is discussed in Action (d).

1.3.5 Design Document Updates

Design Document changes 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 1.14) process is used to control design document changes. Other update processes include UFSAR (Section 1.9), design basis document (Section 1.16), and station procedures (Action (b)). 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 Technical Information Program (VETIP) (Section 1.13) program.

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

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

In each of the remaining sections to Action (a), reference to the process description in Appendix II is made. The process descriptions in Appendix II represent important common elements, as prescribed in the corporate procedures. The following sections supplement Appendix II by describing specific process differences or enhancements made at Braidwood Station.

1.4 Work Control Process

The work control process at Braidwood 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 Action Request (AR). ARs are used to resolve documented problems with systems, structures, and components (SSC) in the plant. Most work at Braidwood, including the initiation of ARs, is processed using the Electronic Work Control System (EWCS). (See Appendix II, Process 15.)

Operators may identify problems with plant equipment while making rounds in the plant. These problems are discussed with the Licensed Shift Supervisor (LSS); ARs are initiated to resolve these problems. A determination is made by the LSS 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 important to safety has been identified as being degraded, an operability assessment is performed.

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 identify any design bases issues. ARs identified by the screening committee as affecting the design bases are sent to the work analysts to coordinate and obtain the requisite engineering input. The use of the screening committee is an improvement initiative to better control the work that may affect the design bases. It ensures that any work affecting the design bases is identified, processed through engineering, and a design change is implemented as required.

The maintenance departments have procedures to control how they perform work. Any work identified by the committee as affecting the design bases is processed and coordinated by a work analyst. The work analyst initiates Engineering Requests (ER) if the design bases is affected. The resolution of these ERs by engineering could result in issuing a modification or exempt change, a temporary alteration, setpoint/scaling request, etc. or it could be a simple clarification response. The work analyst takes this information and uses it in the preparation of the work package.

Part of the work analyst's responsibility is to perform a verification walkdown. This walkdown is performed prior to starting the work package. The performance of this walkdown provides the analyst the opportunity to confirm the configuration of the plant with that provided in the design documents. Any discrepancies between the plant configuration and the design documents are brought to the attention of the engineering department to resolve any document/plant discrepancies by the appropriate change mechanism.

Technical Specification surveillances are controlled through the work control process. This work is performed using controlled procedures. Any discrepancies are reported on a Problem Identification Form (PIF) in accordance with Integrated Problem Reporting (IRP) process described in Action (d).

Replacement parts are evaluated by materials engineering personnel under a like-for-like replacement process in accordance with approved 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. Maintenance verifications are incorporated into the work package instructions through the work control process. Any discrepancies during execution of these verifications require a PIF to be initiated to document the condition.

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. The 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 ISI, IST, Equipment Qualification, etc

1.5 Design Change Process

The Design Change Process at Braidwood is accomplished through the activities described in Appendix II, Processes 2, 3, 4, and 5. The following 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

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 design change starts with the initiation of an Engineering Request (ER). An ER can be requested by anyone on site, however most originate from ARs or are generated within the engineering group. 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 screening/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, a technical and On-Site review is 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 Braidwood 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, several specific areas provide for independent reviews against the design bases. The first area is the Engineering Change Notice which is used to develop the detailed design. Each ECN goes through an interdisciplinary review process, an independent reviewer, and an approver. 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 cannot be installed within tolerance, or an alternate design configuration is required, a Field Change Request (FCR) is generated to evaluate the differences. All FCRs are subject to reviews and approvals by engineering. Additional engineering calculations and 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 which maintain form, fit, and function and require minimal engineering effort. Exempt changes cannot be used for any change that is determined to be 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, including reviews of the design bases and 50.59 evaluations. The mechanics of the exempt changes are similar to the plant modification changes as described above. Engineering Change Notices (ECNs) are required for all safety-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 (SSC) which satisfy established criteria. Temp Alts are not to be used to circumvent the process for permanent plant design changes. The Temporary Alteration 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. The engineer for a Temp Alt assembles the package by marking up copies of the affected design documents, performing a 50.59 screening/review and preparing other pertinent documentation. Temp Alts are independently reviewed by a qualified engineer and the onsite review group. Approved temporary alterations are required to be updated on the Critical Control Room Drawings (CCRD).

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 is responsible for initiating and reviewing the installation of Temp Alts. Temp Alts are installed using a work request and independently verified to ensure the changes are correctly installed. The shift manager is also responsible for ensuring any post modification testing identified by engineering is complete

Temporary Alteration removal is authorized by the shift manager. Temp Alts are not installed without specific plans and criteria for removal. The removal is performed by using the work request 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 Technical Evaluations (Parts and Material Replacement Process)

Parts and 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 units and includes the following objectives:

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

The process is controlled by a Corporate Nuclear Engineering Procedure (NEP) and site procedures. 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 determinati is made that 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 identifi d 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 correct procedures for continuation of the process, i.e., Exempt Change, Modification, etc. 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 the obtainment of 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 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. 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 Inspections (ANI).

The process is audited bi-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 50 59 review screening and Safety Evaluation process at Braidwood is controlled by a Braidwood Administrative Procedure or the corporate Nuclear Engineering Procedure (NEP) The processes are similar in scope and purpose Differences in the procedures have not led to significant differences in 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 After a safety evaluation has been completed, a copy must be sent to the Site Engineering Department, where it is entered into the Safety Evaluation log Copies of all safety evaluations are sent for off-site review (See Appendix II, Process 13) A Peer Group comprised of members from all six ComEd sites reviewed the 50.59 process and identified opportunities for improvement In response, ComEd has proposed a standardized process to be implemented in 1997 The new standardized process will apply to all of ComEd's nuclear stations and replaces all existing processes now in effect

1.7 UFSAR Update

Byron and Braidwood share a common UFSAR update process because they share a common UFSAR The purpose of the UFSAR Update process is to ensure that the UFSAR reflects the current plant status Anyone at Byron or Braidwood can identify a potential UFSAR change and complete an UFSAR change form All proposed UFSAR changes are processed through the UFSAR Coordinator who is part of the Engineering Department Additional major activities within the scope of the process are:

- Plant Modifications
- Setpoint Changes
- Drawing Changes
- Procedure Changes

The UFSAR Coordinator has the responsibility to ensure that the update package is complete, including the 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 the supervisor of the affected department. UFSAR changes are considered part of the UFSAR after the change package is approved and reviewed by management. All proposed changes must be approved by both Byron and Braidwood.

The UFSAR coordinator determines which approved changes will be reflected in the UFSAR update submittal to the NRC. 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.

1.8 Setpoint Change Control Process

The Setpoint/Scaling Change Request (SSCR) process controls the alteration of the existing design or function of a system or component by increasing, decreasing, or removing an existing setpoint. The SSC control program assures that any changes to setpoint values or scaling changes are initiated, analyzed, controlled, and documented in an approved manner. Examples of components to which the SSC 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 set point 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 any 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. After review and approval by the Onsite 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)

1.9 Out Of Service/Return to Service Process (OOS)

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 Braidwood's Electronic Work Control System (EWCS). The main points of the process are:

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 any impacts in the area of Technical Specification (Tech Spec), Primary/Secondary 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 unit.
5. A SRO 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. All cards are hung and then independently verified apart-in-action unless waived by the Unit Supervisor.
7. 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.

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 Technical Specifications and administrative requirement issues.
4. All equipment is repositioned and OOS cards are removed with independently verification apart-in-action unless waived by Unit Supervisor.
5. The Unit Supervisor reviews the RTS to restore Safe Shutdown Paths and to ensure all actions are properly completed
6. A SRO licensed operator will oversee restoration of any Fire Protection requirements.

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, Primary/Secondary Containment impact, 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 removed for RTS, two operators are normally assigned to perform independent verification apart-in-action. 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, equipment is maintained in the correct OOS position, and that the Control Room Simulator adequately reflects the impact of the OOS on the configuration of the plant.

1.10 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 Braidwood's commitment to NRC Generic Letters 83-28 and 90-03, all vendors supplying critical safety related components are recontacted 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 as follows:

The manual or vendor information is logged and tracked by the EWCS or other database.

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

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

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.

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. (See Appendix II, Process 14)

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

The Document Change Process at Braidwood Station is controlled by a NEP/Site Appendix for changes of the station drawings. Plant drawings have been segregated into Critical Control Room

Drawings (CCRD) 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 best drawings available showing the plant as-built information. Schedules based upon the operational date of a design change are prepared for drawings which require updates.

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

1.12 Operability Determination Process

Operability determinations are performed for degraded or nonconforming conditions which call into question the capability of a system, structure, or component (SSC) to perform its specified function(s) as required by the Technical Specifications or UFSAR. Station procedures address the detailed process, which is based on the guidance provided in NRC Generic Letter 91-18.

When an operability issue is identified, Operations expeditiously performs an issue screening. Completion of the issue screening will determine whether the SSC is (1) operable with no concerns; (2) inoperable; or (3) operable with potential concerns. The last determination requires an evaluation to be performed by Engineering.

Operability evaluations are performed by knowledgeable, qualified engineers using detailed guidance. Completion of the operability evaluation will determine whether (1) compensatory actions are required to maintain functionality, and/or (2) corrective actions are required to restore full qualification. The completed operability evaluation is reviewed by Engineering and the Operations Shift Manager

Operability determinations with open compensatory or corrective actions are monitored to ensure satisfactory progress toward closure, and periodically are assessed in the aggregate for any adverse effect on plant operation.

1.13 Computer Software Revisions

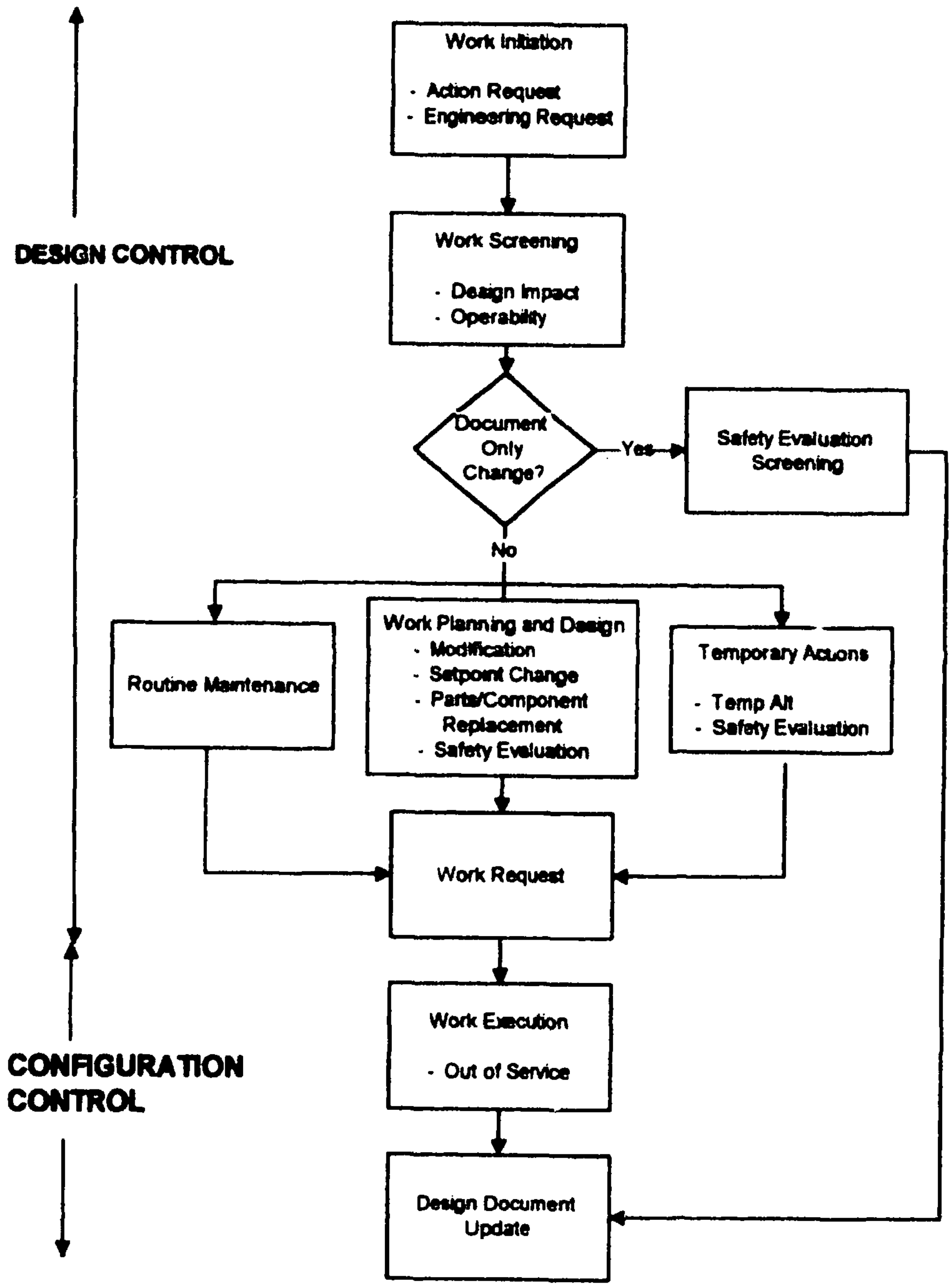
The Braidwood Computer Software revision program applies to software that is safety related, used to perform controlled engineering work. Once a need to develop or revise Engineering Software has been identified, a Software Activity Request is generated to describe the circumstances and identify the 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)

1.14 Summary

The programs and processes used at Braidwood to maintain configuration control are developed and implemented in a manner consistent with industry standards. Audits, surveillances, and assessments have been conducted by the responsible line organizations, by Site Quality Verification, and by third parties; and these have not identified any deficiencies representing significant breakdown of the quality assurance program. Where deficiencies in process and procedures have been identified, corrective action has been implemented to address and prevent recurrence of these deficiencies.

The implementation of these processes provide reasonable assurance that the configuration and performance of Braidwood Station is consistent with its design bases.

Typical Flow of Design and Configuration Control Processes



2.0 Action (b)

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

2.1 Introduction

Braidwood Station implements a comprehensive procedure preparation and revision process, in accordance with applicable license and Quality Assurance requirements, which provides reasonable assurance that appropriate design bases requirements are translated into operating, maintenance, and test procedures. The rationale for concluding that design bases requirements have been translated accurately into operating, testing, and maintenance procedures is:

- Original plant 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 plant hardware prior to plant startup.
- Subsequent to startup, some procedures have been revised and new procedures have been prepared in accordance with applicable station administrative procedures which implement Quality Assurance requirements. When the plant was licensed, the NRC concluded that these original administrative 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 in the development 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 accurately into these procedures.
- The number and type of procedure deficiencies, as obtained from recent Problem Identification Form (PIF) reviews suggests no significant deviations from design bases or adverse trends in this area.
- Audits and inspections by both ComEd and external agencies have concluded, in general, that the procedure control and revision process provides reasonable assurance that design bases information has been translated accurately into operating, maintenance, and testing procedures. In those few cases where problems were identified, appropriate corrective actions were implemented.

Each of these elements is discussed in more detail below

2.2 Consistency of Original Station Procedures with Design Bases

Original plant testing, maintenance, and operations procedures were prepared prior to startup by the NSSS vendor, Architect Engineer, and ComEd. Other plant operating experience, vendor equipment requirements, and design basis were all considered in the preparation of these procedures. Many of the Braidwood procedures were based on the Byron procedures (Byron and Braidwood were licensed under the replicate plant option of 10 CFR 50 Appendix N), which had been in use for a period of approximately eighteen months prior to Braidwood's startup. Many of Braidwood's procedures were implemented during testing and other pre-startup activities, such as instrument and control system calibration, system acceptance testing, power ascension, and trip response testing. Formal verification efforts were conducted to ensure the adequacy of the original procedures and their conformance with the licensing and design bases. Two significant examples are (1) a review of pre-operational and startup testing conducted by the ComEd Quality Assurance organization for all safety-related systems, to ensure that functions described in the UFSAR 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 results of these and other reviews were relied upon by the NRC when it concluded that Braidwood's procedures were acceptable at the time of issuance of the operating license.

2.3 Procedure Preparation and Revision Process

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.

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

The procedure preparation and revision processes include the following elements, which 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
- Technical Review or Onsite Review
- Station Manager Review (procedures which require Onsite Review)
- Review by Plant Operation Review Committee (PORC) (selected procedures)
- Verification and/or Validation
- Commitment Preservation

As a minimum, all new procedures and procedure changes are required to undergo a 10 CFR 50.59 screening and technical review prior to approval. For procedures which have a possible safety impact, and/or as required by NRC regulation, additional reviews are performed. These additional reviews may consist of Onsite Review, Station Manager review, and PORC review, as appropriate.

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, including the design bases requirements contained in the UFSAR. Personnel who perform this screening must meet the qualification requirements specified in station procedures for minimum education, training, and power plant experience. These requirements help to assure that procedure preparers and reviewers have the necessary knowledge of design bases information.

2.3.2 Technical Review

Technical reviews are performed on all newly prepared or revised testing, maintenance, and operating procedures to confirm technical adequacy and compatibility with existing plant design and operation. Technical reviews are performed by personnel knowledgeable in the subject matter and who meet the applicable experience requirements of Sections 4.2 and 4.4 of ANSI N18.1-1971. 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. Detailed review guidelines are used by technical reviewers. Attributes reviewed include:

- Review of applicable plant drawings and vendor information,
- Determination of whether the procedure or revision addresses lessons learned and station commitments,
- Review of the UFSAR, including pending changes, and
- Impact on systems, other procedures, other programs (EQ, ISI/IST, etc.), personnel safety, safety-related equipment, and plant or control room operations.

2.3.3 Onsite Review

A subset of plant procedures require Onsite Review as based on NRC Regulatory Guide 1.33, Revision 2. This requirement is formalized in the ComEd QA Topical Report. Procedural guidance is specified for Onsite Review, and the following critical attributes are specified for key attention.

- Fulfillment of Technical Specification requirements,
- Fulfillment of UFSAR requirements and commitments,
- Identification and resolution of Safety issues,
- Fulfillment of station commitments to NRC, INPO, and other agencies.

Onsite reviewers are also required to be qualified per ANSI N18.1 as described above for technical reviewers.

2.3.4 Review by Plant Operation Review Committee (PORC)

Procedures which require an Onsite Review are also reviewed for safety impact to the operation of the plant. If a potentially significant safety impact is identified, the procedure may be reviewed by the Plant Operation Review Committee (PORC). The PORC consists of senior station management (operations, maintenance, radiation protection, chemistry, engineering, and other departments) who provide multiple perspectives on the adequacy of the procedure under review. Although the focus of this review is safety, it provides an opportunity for knowledgeable senior station managers to identify any concerns regarding the accuracy of translation of design bases information.

2.3.5 Station Manager Review

In accordance with the requirements of the ComEd QA Topical Report, those procedures which require Onsite Review are also required to be approved by the Station Manager. In providing that approval the Station Manager confirms that appropriate participants were selected for the Onsite Review; that the review was in sufficient depth; and that the findings and recommendations are reasonable.

2.3.6 Verification and/or Validation

New procedures and procedure revisions may require verification of key attributes as deemed necessary by the appropriate plant Supervisor. The following are examples of attributes which may be subject to verification:

- Format conforms to the Writers Guide,
- Procedure meets its stated purpose,
- Procedure includes adequate steps and information to perform the intended function,
- Equipment numbers are identical to labels in the field,
- Required equipment is specified,
- Sequencing of actions is appropriate, and
- Expected plant/equipment response is reasonable.

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 by using the plant simulator is considered during procedure review and is dependent upon the type of procedure or change and safety significance. For example, changes to Emergency Operating Procedures (EOPs) require validation using the plant simulator.

2.3.7 Commitment Preservation

The procedure Writers Guide includes a key requirement which helps to assure that procedures are consistent with design bases; namely, the requirement to specifically identify those procedure steps which fulfill commitments, and to include the commitment in the procedure's "Reference" section. No such steps may be changed or deleted without consideration of the source of the commitment.

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, and 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, Braidwood has experienced the following unplanned events which demonstrated that its safety systems responded appropriately, and in which procedures were effectively utilized to respond to the events:

- 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 a spurious main steam isolation signal caused the Unit 1 reactor to trip due to low SG water 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 1995, the Unit 1 reactor was tripped manually when a test relay failed during control rod drop testing. All rods fully inserted as designed.
- In 1996 Unit 2 experienced a Loss of Offsite Power due to the loss of both Unit 2 system auxiliary transformers. All automatic bus transfers functioned as designed. Unit 2 remained stable at 100% power throughout the event.

2.5 Programs Which Verify Procedure Consistency

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 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 system, 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

1. Does the difference appear to adversely impact a system or component explicitly listed in the TS?
2. Does the item appear to compromise the capability of a system or component to perform as described in the UFSAR?
3. Does the difference appear to adversely impact any applicable licensing commitments?

As described in Action (a) of this response, ComEd formed a Process Improvement Team to develop an improved UFSAR update process for all Stations. Also, as reported in a meeting at NRC Region III on 8/26/96, ComEd will perform conformance reviews of additional UFSAR sections at all its Stations. The extent, schedule, and relationship of these reviews to other ongoing initiatives is presently under evaluation

2.6 PIF Trends and Data Analysis

A review of Problem Identification Forms (PIFs) from 1992 to present was performed and the results were trended. No programmatic deficiencies were identified in the area of procedure consistency with design basis. Where problems were noted, effective corrective actions were taken to resolve the problem and prevent recurrence

A potentially significant deficiency was identified in February, 1994, when batteries which provide backup power to control room HVAC dampers were found to be inoperable. As described in NRC Inspection Reports 94008 and 94015, this event resulted from the failure to translate available design information into appropriate surveillance and preventive maintenance procedures. As detailed in ComEd's response to the Inspection Reports, significant corrective actions were taken to prevent recurrence, including reviews of other systems to verify that proper surveillance and preventive maintenance are being implemented, and improved training for engineers which emphasized the need to ensure that design information is appropriately incorporated into plant procedures. Those actions have proven effective, as indicated by the absence of similar events

2.7 Inspection and Audit Results

A large number of audits and inspections by both ComEd and external agencies have been performed over the years to confirm the procedure update and revision process is correctly implemented and to confirm the adequacy of prepared procedures. Vertical slice audits (using methodologies similar to that of the NRC SSFIs) have also been performed on select plant systems. These audits have confirmed procedure adequacy on a sampling basis. Major audits and inspections include:

- A ComEd Nuclear Quality Programs (NQP) inspection of the Auxiliary Power System in 1992.
- NRC Inspection 93009, an Electrical Distribution System Functional Inspection (EDSFI) in 1993.
- A ComEd Site Quality Verification (SQV) inspection of the Auxiliary Feedwater System in 1993
- A ComEd Site Quality Verification (SQV) inspection of the Control Room Ventilation System in 1994.
- A ComEd Site Quality Verification (SQV) audit of Site Engineering Activities in 1995
- NRC Inspection 95003, an inspection of the Motor-Operated Valve (MOV) program in 1995.

A brief summary of the audit scope and results, in the area of procedures, is provided below.

ComEd NQP Inspection Of Auxiliary Power System

Focused on operability and readiness of Auxiliary Power System, including conformance to UFSAR descriptions and Tech Spec surveillance requirements. Found generally good consistency between procedures and licensing and design bases. A deficiency was found and corrected in the use of potentially non-conservative acceptance criteria for MOV thermal overload testing. An enhancement was identified and implemented in the position verification surveillance procedure for support cooling water valves.

ComEd SQV Inspection Of Auxiliary Feedwater System

Focused on Auxiliary feedwater system operability and material condition, including lubrication, post maintenance testing, and calibration. Concluded that system was in generally good condition and capable of performing all its required functions. A deficiency was found and corrected in that the lubrication oil check frequency of one (1) of four (4) aux feedwater auxiliary lube oil pumps had been set to a nonconservatively long interval

NRC EDSFI

Concluded that electrical distribution systems were designed with ample margin and that the design had been correctly translated into operating, maintenance, and testing procedures. Described surveillance procedures as comprehensive, user-friendly, and containing clearly defined acceptance criteria; and considered them to be a strength

ComEd SQV Inspection Of Control Room Ventilation System

Focused on all aspects of control room ventilation system readiness, including surveillances, preventive maintenance, and incorporation of vendor information. Two separate reviews were performed in the area of surveillances. First, the Technical Specification, UFSAR, and the station commitment database were reviewed, and it was verified that all the testing requirements were met by the station programs and procedures. Second, a review was performed of the original pre-operational tests to identify all of the functional tests which were performed and which equipment was affected. It was confirmed that all the original functional tests were being performed as part of the ongoing surveillance program, with a small number of exceptions which were then evaluated and incorporated into the program as appropriate. In addition, several inconsistencies were noted and resolved between the vendors' recommended preventive maintenance frequencies and the station's surveillance procedures.

ComEd SQV Audit Of Site Engineering

Focused on several aspects of Site Engineering's activities, including design modifications, calculations, and engineering interfaces. Concluded that Site Engineering was generally doing a good job of engineering plant changes and accessing and utilizing design basis information, and that appropriate procedure changes were being developed and implemented for plant modifications.

NRC MOV Inspection

Concluded that MOV diagnostic test procedures were thorough, incorporated pertinent sources of equipment inaccuracy, included margin for degradation, and extrapolated the minimum required thrust to design basis conditions; and that plans for periodic verification of MOV design basis capability were generally acceptable

2.8 Summary

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.

3.0 Action (c)

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

3.1 Introduction

The bases for Braidwood Station's conclusion that the configuration and performance of its structures, systems, and components (SSCs) are consistent with their design bases can be summarized as follows. When Braidwood was licensed to operate by the NRC, that license was supported, in part, by ComEd's certification that Braidwood's safety-related SSCs were configured in accordance with, and will conform with the plant's design bases. Since then, Braidwood 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 and procedures which were designed to preserve the configuration and performance of SSCs in conformance with their design bases. These processes and procedures have been described in Actions (a) and (b) 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 demonstrates conformance of the SSCs with their design bases. A large body of data about SSCs configurations also has been developed over the years as various SSCs have been 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 actions have been taken. These elements are discussed in more detail below.

3.2 Initial Determination That Configuration And Performance Were Consistent With The Design Bases

As part of the initial licensing process, by letter of October 1, 1986, ComEd certified that Braidwood 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 (Westinghouse), the architect engineer (Sargent and Lundy), and ComEd (and their extensive experience in applying these practices to other successful projects, such as Byron); (2) the contributory roles of extensive ComEd quality assurance audits and third party reviews; and (3) extensive inspections and scrutiny by the NRC. Braidwood and Byron were constructed as replicate plants under the same procedures and practices. Consequently, Braidwood received the benefit of the reviews and inspections conducted on Byron (which, for example, included a special NRC Construction Assessment Inspection.) The product of the combined inspection and assessment efforts on

Byron and Braidwood provided a high level of assurance that Braidwood's initial configuration and performance were consistent with the design bases.

Similarly, by letter of October 3, 1986, ComEd certified that the Technical Specifications accurately reflected the as-built plant and the FSAR. As previously described in the response to Action (b), these certifications included detailed verifications of the initial complement of plant operating, maintenance, and testing procedures for consistency with the licensing documents and the physical plant.

3.3 Preservation Of Configuration And Performance Consistent With The Design Bases

Since initial plant startup, SSC configuration and performance have been maintained consistent with their design bases through the implementation of p:ograms, processes, and procedures which 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 Action (a) 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 Action (b) of this response. Ongoing plant performance is monitored through operator and other plant personnel actions as described in more detail in Section 3.4.

In 1996, following a series of events which included mispositioning of some valves and the failure to follow a number of administrative provisions of the Out of Service process, Braidwood took a number of actions to improve its performance in the area of operational configuration control. In addition to actions directed at the specific events, broad corrective actions were implemented. There included the re-performance and verification of valve and equipment lineups for important plant systems, and the implementation of an enhanced Out of Service process. In addition, management clarified and reinforced expectations for Operators, Work Analysts, and the entire Site population, regarding procedural adherence, the Out of Service process, and operational configuration control

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

System Engineers are also expected to perform regular plant walkdowns of their systems to validate system configuration and to identify any system deficiencies. Informal walkdowns are done frequently on assigned systems and components. A formal, documented walkdown of each assigned system is accomplished on a quarterly basis, which includes interviews and discussions with shift operating personnel concerning system and component problems.

The purpose of the walkdown is to identify (prior to failure) problems which could affect system operation, and problems already identified which may not have received the proper priority for resolution. System Engineers look for any unauthorized temporary alteration and take actions to rectify them. They use their expertise and knowledge of the system design bases to aid in finding of potential problems related to configuration control and/or equipment performance. Additionally, formal reviews of system performance (System Readiness Reviews) are conducted on a routine basis to maintain system performance at the desired level.

3.4.2 Surveillance Testing

A comprehensive program of testing SSCs 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 and the performance of mechanical components to ensure reliable performance upon demand throughout the plant lifetime.

Periodic surveillance testing is performed in accordance with Technical Specification requirements and In-service Testing Program requirements. The testing procedures verify that critical system performance parameters are satisfied during system operation. Testing discrepancies require evaluation for operability, and may be evaluated for root cause and corrective action determination via the station corrective action (PIF) process, as necessary.

Braidwood Station conducted an extensive preoperational test program which was based on the guidance of Regulatory Guide 1.68. As described previously, reviews were conducted which confirmed (1) that all safety-related system functions described in the UFSAR 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 (on a sampling basis) all important functions tested by the preoperational tests have been carried forward in the ongoing surveillance testing programs.

The Braidwood surveillance testing programs have been audited periodically for conformance to the current Technical Specifications, which are based on the Westinghouse Standard Technical Specifications, NUREG 0452 Draft Rev. 5. No significant deviations have been found. Braidwood has begun the process of conversion to the Improved Technical Specifications (NUREG 1431). 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. 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 plant work control process previously described includes requirements for review of all work packages prior to issue for work, and specification of any required post maintenance testing. This process ensures that the work is done properly and that the equipment conforms to applicable specifications.

The plant modification process requires 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 Action (a) of this response. Construction tests are performed to ensure that installation work is performed correctly and in accordance with the codes and standards governing the work. Modification Testing ensures that the plant change performs as expected when connected into the plant systems. Operability Testing is performed to ensure that the modified equipment will meet the surveillance requirements in the Technical Specifications. The testing requirements are implemented 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 modified installation remains consistent with the design bases. Audits and inspections which have confirmed the effectiveness of this testing are discussed in Section 3.7 of this Action (c) response.

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 function. Routine monitoring and assessment of the performance of SSCs that contribute most significantly to plant safety provides added assurance these SSCs function when and as required; and therefore, that SSC performance is consistent with the design bases.

The Maintenance Rule implementation and compliance program at Braidwood 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. Any deviations from NUMARC 93-01 guidelines are identified in the Maintenance Rule Implementation Procedure.

3.5 Operating Experience

Performance of the plant as expected provides additional verification that SSCs configuration and performance have been maintained consistent with the design bases. Braidwood's successful response to equipment failures and transients provides assurance that the configuration and performance of key SSCs has been maintained. 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 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 1995, the Unit 1 reactor was tripped manually when a test relay failed during control rod drop testing. All rods fully inserted as designed.

- In 1996 Unit 2 experienced a Loss of Offsite Power due to the loss of both Unit 2 system auxiliary transformers. All automatic bus transfers functioned as designed. Unit 2 remained stable at 100% power throughout the event.

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:

- Assembling design and licensing information and improving its accessibility
- Revising or establishing more 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

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 architect engineer and developed in-house engineering capability. As part of this effort, ComEd acquired the majority of calculations used by the NSSS supplier and the architect engineer in the design of SSCs important to safety (and in subsequent modifications.) For Byron and Braidwood, which were designed, constructed, and licensed as replicate plants, more than 165,000 calculations have been acquired. These have been indexed in ComEd's Electronic Work Control System (EWCS) and are available at the Byron and Braidwood Stations.

During the same time period, ComEd has made the UFSAR 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 Braidwood's ability to maintain plant configuration and performance on an ongoing basis.

3.6.2. Revising Or Establishing More Specific Calculations Which Implement The Design Bases, And Verifying That Plant Configuration And Performance Is Consistent With Design Information

- Instrument Setpoints

  ComEd developed a standard instrument database, along with a standard methodology for performing the supporting setpoint calculations. For Braidwood, calculations were verified or re-performed for reactor protection and engineered safety features actuation setpoints, in order to ensure consistency between actual plant setpoints and channel accuracies and the design bases.

- Fuses

  Similar to instrument setpoints, ComEd developed a standard fuse database, along with a standard engineering process for fuse selection in design applications. For Braidwood, the majority of safety related fuses have been walked-down in the plant, to provide additional assurance of conformance with the design bases. Discrepancies have been resolved as necessary.

- Motor Operated Valves (MOVs) Program

  To meet the requirements of GL 89-10, ComEd documented the design bases for safety related MOVs, reconstituted calculations, established performance requirements, and performed comprehensive static and dynamic testing; and has adjusted MOV setpoints, modified equipment, and revised operating and maintenance practices as necessary to ensure that all safety related MOVs will reliably perform their intended function under design bases conditions. The NRC performed inspections which confirmed that the Braidwood program has met that objective.

- Degraded Voltage Setpoints

New degraded voltage relay setpoints have been determined for Braidwood. This analysis was 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 also been performed on motor operated valves, essential 120VAC, and contactor circuits. These supplementary calculations now serve as part of the plant design bases.

- 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 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 system, 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:

1. Does the difference appear to adversely impact a system or component explicitly listed in the TS?
2. Does the item appear to compromise the capability of a system or component to perform as described in the UFSAR?
3. Does the difference appear to adversely impact any applicable licensing commitments?

As described in Action (a) of this response, ComEd formed a Process Improvement Team to develop an improved UFSAR update process for all Stations. Also, as discussed in a meeting at NRC Region III on August 26, 1996, ComEd will perform conformance reviews of additional UFSAR sections at all its Stations. The extent, schedule, and relationship of these reviews to other ongoing initiatives is presently under evaluation.

- Review of Appendix R Safe Shutdown Analysis

In 1994, a review of the Safe Shutdown Analysis implementing 10 CFR 50 Appendix R was initiated as a part of the station's response to NRC Generic Letter 92-08 "Thermo-Lag 330-1 Fire Barrier" issues. The intent of the review was to determine if some Thermo-Lag fire barrier issues could be resolved by re-analysis of Appendix R implementation at the site. During the course of the review, some analysis errors and discrepancies between analysis and actual plant installations were identified, which were resolved in accordance with established procedures for assessment of operability, reporting, and restoration of conformance. An independent assessment of the Byron and Braidwood Safe Shutdown Analyses was performed by Corporate ComEd and a non-ComEd consultant. This independent assessment recommended further review in selected areas. This review is ongoing and is expected to continue into 1997. Byron and Braidwood are confident that any remaining deficiencies, yet to be discovered, will be found and corrected.

- Improved Technical Specifications

The Byron/Braidwood 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 Braidwood Station to modify the NUREG document into a Station-specific document. The design and licensing bases of the station licensing SER, UFSAR, other design documents, and safety analyses were utilized in developing the Station-specific ITS document. Each provision of the current Technical Specifications (CTS) was equated to a provision of the NUREG document or a discussion was provided to justify any changes.

As each ITS section and the associated bases were developed, it 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. This included the development and review of a bases section that consists of sections discussing the background for the specification, applicable safety analyses, and applicability of the LCO, the LCO, its actions, 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 had completed the technical reviews, a final Integrated On-Site Review of the project was conducted. After completion of the Integrated On-Site Review, a Corporate Off-Site Review was conducted. At the completion of the onsite approval process, the Byron/Braidwood Improved Technical Specifications were submitted for NRC review. The NRC's review is expected to be completed in 1997.

3.6.3 Establishing Special Monitoring Programs To Confirm Conformance With Specific Aspects Of Design On An Ongoing Basis

- Emergency Diesel Generator (EDG) Reliability Program:

  ComEd implemented an EDG Reliability Program in 1993. (Regulatory Guide 1.108 controls applied previously.) 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 ensures conformance with the design and licensing bases by maintaining and monitoring EDG reliability over time for assurance that the selected target EDG reliability levels are being achieved.

  The EDG Reliability Program includes requirements for 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).

  Braidwood has maintained a high level of EDG reliability since the implementation of the program.

- NRC Generic Letter 89-13:

  NRC Generic Letter 89-13 required Braidwood Station to confirm that essential service water systems would perform their intended functions in accordance with applicable design bases. System design, testing, and operation were reviewed; and no discrepancies were found which would call into question the capability of SSCs to perform their design bases functions. A program was implemented to monitor overall system performance on an ongoing basis, as required by the Generic Letter. Implementation of this program provides added assurance that the essential service waters system will function in accordance with the design bases.

3.7 Audits And Inspections Of Configuration And Performance

Braidwood has undergone an extensive series of audits and inspections which have provided additional assurance that the SSC configuration and performance are consistent with the design bases. These include self-assessments by the responsible line organizations; Quality Assurance audits; third party audits and assessments; ISEG evaluations; and NRC inspections. "Vertical slice" type inspections have been performed on selected plant systems. These include:

- A ComEd Nuclear Quality Programs inspection of the Auxiliary Power System in 1992.
- A ComEd Station Quality Verification inspection of the Auxiliary Feedwater system in 1993.
- NRC Inspection 93009, an Electrical Distribution System Functional Inspection (EDSFI) in 1993.
- A ComEd Station Quality Verification inspection of the Control Room Ventilation System in 1994.

A brief summary of the audit scope and results, in the area of configuration and performance of SSCs, is provided below.

ComEd NQP Inspection Of Auxiliary Power System

Focused on operability and readiness of Auxiliary Power System, including conformance to UFSAR descriptions and Tech Spec surveillance requirements. Concluded that the system remained capable of performing its design bases functions.

ComEd SQV Inspection Of Auxiliary Feedwater System

Focused on Auxiliary feedwater system operability and material condition, including lubrication, post maintenance testing, and calibration. Concluded that system was in generally good condition and capable of performing all its required functions.

NRC EDSFI

Concluded that electrical distribution systems were designed with ample margin and were capable of performing their design bases functions. Reviewed thirteen (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.

ComEd SQV Inspection Of Control Room Ventilation System

Focused on all aspects of control room ventilation system readiness, including surveillances, preventive maintenance, and incorporation of vendor information. 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. Concluded that the system remained capable of performing its design bases functions.

3.8 PIF Trends And Data Analysis

A review of Problem Identification Forms (PIFs) from 1992 to the present was performed. With few exceptions, no programmatic deficiencies were identified in the area of configuration and performance consistency with the design bases. Where problems were noted, effective corrective actions were taken to resolve the problem and prevent recurrence. Adverse trends which were identified and acted upon are discussed as follows:

1. Unauthorized Temporary Alterations

In July, 1995, a temporary fan was found to be attached to a safety-related battery room fire damper, without formal evaluation under the Temporary Alteration process. Corrective actions included a walkdown of all safety related systems, which identified 23 potential unauthorized temporary alterations. Evaluations determined that there were no immediate operability or safety significant issues associated with these items; and they were subsequently dispositioned by removal or by processing them as design changes. Broader corrective actions included training of Operators in support systems and their effect on operability, and training of a broad segment of the Station population (Operating, Engineering, Maintenance, Chemistry, Station Laborers, and Security) in how to recognize Temporary Alterations and the provisions of the Braidwood Temporary Alteration procedure. Corporate-wide improvements to the Temporary Alteration process are discussed in Appendix II of this response.

2. Modification Testing Not Timely

In 1994, a trend review identified a relatively high number of "open" modifications for which installation and testing had not been completed in a timely manner. Corrective actions were developed for the modification process, but they were not fully implemented until October, 1996. In August, 1996, following the identification of a large number of open modifications at Zion Station, all ComEd Sites initiated additional reviews for similar items. In September, 1996, Braidwood experienced the failure of certain ventilation barriers (roll-up doors) to meet their design function. Subsequently it was determined that the modification testing for the roll-up doors had not been completed. Corrective actions included the identification, review, and safety evaluation of all other design changes which had been in the installation and testing phase for an extended period of time, and the development of an expedited closure plan; along with procedure changes which require monthly management reviews to ensure that design changes are progressing satisfactorily to completion.

3.9 Summary

Based on the formal certification of conformance which was part of the original licensing process; the preoperational verification and testing activities; the ongoing verifications 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 inspections, including the resolution of identified problems and the implementation of improvement initiatives; there is reasonable assurance that Braidwood's SSCs 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 Braidwood Station to identify problems, to determine the root cause and the extent of the problems identified, and to report problems to the NRC It also describes the processes used to resolve problems identified through implementation of appropriate corrective actions, including actions to prevent recurrence In addition to the Integrated Reporting Program (IRP), the corrective action program which is generally applicable during routine operations, this section also addresses the identification and correction of problems through special programs, targeted reviews, audits, and inspections.

4.2 Integrated Reporting Program (IRP)

The Integrated Reporting Program (IRP) provides a consistent method for identifying problems, investigating such problems through a controlled process, controlling and tracking corrective actions, and reporting problems to the NRC. The IRP also includes requirements for trending of problems to address any programmatic concerns and identify improvements where necessary

The IRP applies to conditions adverse to quality which are related to design bases conformance The IRP ensures that design concerns are formally documented and evaluated. Identified conditions adverse to quality are assessed to determine their impact upon operability in accordance with NRC Generic Letter 91-18. Identified conditions adverse to quality are assigned a priority for timely corrective action based upon their significance to safety. Station processes require that 50.59 safety evaluations be prepared if the plant intends to operate for a prolonged period with a degraded condition that does not conform with the FSAR. Those conditions determined to be 'significant' by the Event Screening Committee (ESC) using specific screening criteria are subjected to root cause investigations to prevent recurrence of problems. Problems are tracked through implementation of corrective actions and trended to identify those that recur. Station processes are also in place to prevent recurrence of problems, and to identify and correct any generic implications.

The IRP process uses a Problem Identification Form (PIF) for reporting problems. It is available to all station personnel for identification of any problem, condition adverse to quality, or concern The PIF process is described in station procedures that contain necessary forms which are simple to use. This prescribed PIF form is readily available at various locations at the station

The threshold for creating a PIF is very low. Guidance regarding the type of problems to be reported is contained in procedures Moreover, station management aggressively encourages all station personnel, via General Employee Training and/or through reinforcement of expectations, to document any concerns.

The PIF process includes provisions for promptly identifying and evaluating operability and reportability Once a PIF is initiated, it is reviewed by an SRO to determine whether an immediate nuclear safety or operability concern exists If such a concern does exist, the Shift Manager takes appropriate action to place the plant in a safe condition During the review of the PIF, the SRO also uses established guidance to determine the reportability of the problem All new PIFs are reviewed during each shift. If additional input from Engineering is required to confirm the initial determination of operability, then the issue is forwarded to Engineering for an "Operability Evaluation " Station procedures specify time constraints for completing operability evaluations.

All PIFs are reviewed by the Event Screening Committee(ESC) which meets daily (except on weekends and holidays). The ESC evaluates the adequacy of any immediate corrective actions taken prior to their review, determines whether follow-up actions are required, determines whether additional information is required, and classifies the PIF as "significant" or "not significant " For PIFs determined "significant", the ESC assigns an action for root cause investigation to the appropriate station organization using established criteria. The ESC mission is to assure appropriate resources are assigned for a timely investigation of consequential events The ESC uses the combined group knowledge to understand the importance of plant problems, determine which PIFs might have NRC reportability requirements, focus management's attention on significant issues, and review completed consequential events to verify corrective actions are appropriate for the root cause The ESC reaches concurrence on those PIFs which need to be resolved immediately by line managers, and those for which a root cause analysis will be conducted

The station PIF coordinator is responsible for tracking PIFs. Progress in closing out PIFs is monitored through the Nuclear Tracking System (NTS), a computer database. The Station PIF coordinator assigns actions in the NTS when corrective actions or root cause investigations are required. Procedures require that corrective actions to prevent recurrence resulting from root cause investigations be reviewed for effectiveness

4.3 Other Processes Which 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 and are the primary vehicles used to effect repairs and other work on plant equipment. All station personnel may initiate an AR on deficient equipment. The AR is sent to the station's Screening Committee, made up of individuals with knowledge of plant licensing and design bases. The AR is assigned to the appropriate work group and the AR is submitted to Work Control. For modifications, an Engineering Request is generated and assigned to Engineering for processing under the controls of the modification process

4.3.2 Engineering Requests (ER)

An ER is used as a method of requesting assistance from Engineering ERs may be used to provide an evaluation of a potential or existing problem The station procedures provide guidelines on how plant personnel submit technical inquiries, requests for design evaluation, and design change requests to the Engineering Department PIFs are generated for any design bases nonconformances which may be identified during the processing of an ER and are routed to the IRP process as needed

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 described in Action (a) 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 Direction would be issued to restore the physical plant to its design condition if necessary

4.3.4 Operating Experience Reviews (OPEX)

The Operating Experience Feedback (OPEX) information process is described in station procedures The OPEX Administrator is responsible for the screening and distribution of operating experience information, including INPO SOERs and SERs, NRC Information Notices and Generic Letters, Vendor bulletins, etc OPEX procedures specify the internal distribution requirements for routing, review, and actions to be taken for operating experience information Due dates are assigned for responses, and actions are tracked using the Nuclear Tracking System (NTS) Regulatory information involving the NRC is processed by the Regulatory Assurance group Information determined to have vendor manual impact is forwarded to the VETIP Coordinator ERs and/or PIFs may be initiated for issues related to design bases conformance

4.3.5 Operator Work Arounds (OWA)

Operator Work Arounds are 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, or compensatory procedural requirements to perform a task due to degraded equipment An operator work around is identified as a problem and is processed through either the PIF process or the AR process OWAs are assigned a tracking number and prioritized by Operations Department management The item is then assigned to an owner, usually the System Engineer The System Engineer will develop an action plan for resolution of the issue and process an AR (or ER if required for design changes) Open OWAs are reviewed periodically for aggregate impacts by Site management

4.3.6 Technical Alerts

Technical Alerts are a special ComEd form of operating experience feedback related to engineering issues, identifying "lessons learned" at one station and making them available to the other stations The Tech Alert contains detailed information on engineering issues and includes solutions identified and actions needed to address the issue at other locations This program is run by the Corporate Engineering staff

4.3.7 Nuclear Operations Notifications

A Nuclear Operations Notification (NON) notifies other ComEd Nuclear sites of a problem or event which 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 generally published before the event investigation is completed They are posted on a ComEd computer bulletin board so that anyone with a cc:Mail account has immediate access to them At Braidwood, the Event Screening Committee (ESC) may elect to initiate a NON from the PIFs they evaluate daily If it is determined that a NON should be published, it will be prepared by the cognizant station personnel and posted to the cc Mail bulletin board Each NON published by other sites is forwarded to the appropriate station department for information and/or applicability review if the problem appears to be potentially applicable to Braidwood Station

4.3.8 Audits And Evaluations

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

Site Quality Verification (SQV) Audits: Since 1990, the SQV organization has conducted performance-based audits Prior to that, audits were compliance based 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 (N O) procedures and SQV instructions The 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), which is discussed in Section 4.5.1

Independent Safety Engineering Group (ISEG): As with SQV audits, the Independent Safety Engineering Group (ISEG) performs reviews in accordance with Nuclear Oversight procedures and SQV instructions The group function is to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports (LERs), and other operating experience information, including plants of similar design, which may indicate areas for improving unit 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 Deficiencies identified during ISEG reviews are documented on a CAR, which is discussed in Section 4.5.1

Field Monitoring Program: conduct of and preparation for field monitoring activities is also an SQV function Field monitoring activities are intended to focus on adverse or declining performance areas 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, test, or maintenance procedures or sequences Deficiencies are documented on a Field Monitoring Report and a CAR, which is discussed in Section 4 5 1

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 Trends (both positive and negative) are reported to station and Nuclear Oversight Department management via a monthly report The Quality Control (QC) group also trends weaknesses identified during work request reviews and field inspections and provides written reports to management PIFs are initiated when adverse trends are identified

Quality Control Program: The Quality Control (QC) program is controlled by station procedures Discrepant items, such as components, parts, spares, consumables, portable test equipment, and inspection and test procedures that are identified in the field are documented on PIFs

4.3.9 Quality First

The Quality First Program is a program through which Nuclear Operations Division employees and contractors are able to address concerns that are directly and indirectly related to quality and safety Employees and contractors are encouraged to voluntarily raise any concerns they may have in the performance of their jobs through this program In general, individuals who wish to raise potential deficiencies or problems work through their supervisors All supervisors receive guidance in the process and are expected to be sensitive to potential concerns, clarify communications to assure mutual understanding, and act upon potential concerns in a timely manner

ComEd management has high expectations for the entire Nuclear Operations Division when it comes to quality and safety ComEd management also expects supervisors and line management to create an atmosphere in which employees can freely voice concerns The individual raising the concern may request confidentiality and every effort will be made to assure that 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

4.4 Processes Which Determine The 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

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 a PIF has been screened by the Event Screening Committee which meets daily, except on weekends and holidays, to review PIFs initiated since the last meeting. The ESC may assign a root cause investigation based on the severity of the issue as part of the screening/review process. Root Cause investigations, evaluations, and reports are conducted in accordance with station procedures. Root Cause investigation techniques, checklists, and report format are provided in station procedures as well as Root Cause Manuals. Appropriate training is conducted for Root Cause team members

4.4.2 PIF Trending

Currently, a trend coordinator reviews all PIFs, assigns a type code, and provides trend data to individuals and managers as identified or on request. The trending provides substantial intelligence and can identify where additional corrective actions are required.

All PIFs are trended on a periodic basis to identify recurrent human performance issues. The focus is on those PIFs which are not classified as "Significant," but which are important enough to undergo an "Apparent Cause" Evaluation. If a trend of recurring, similar human performance problems, with similar apparent causes is detected, a "Trend PIF" will be generated by the Trend Analyst. This Trend PIF will be automatically classified as "Significant Condition Adverse to Quality," and undergo a Root Cause Investigation.

In addition to the human performance trending that is regularly performed by the Trend Analysts, System and Component Engineers are beginning to trend PIFs in order to identify adverse performance trends of their assigned systems or components. Although this trending program is not yet as formalized as the human performance trending, it is growing in application as experience with the PIF process increases and the PIF database expands to make trending a worthwhile activity for the System and Component Engineers

4.5 Processes Which Identify And Implement Corrective Actions

4.5.1 Corrective Action Records (CARS)

This program is administered by the SQV organization on site. A Corrective Action Record is a stand-alone document used to identify concerns or strengths developed during field monitoring activities, audits, and surveillances. The CAR is used for documenting, reporting, followup, condition closeout and trending. There are four status levels and three levels of significance in the CAR program.

4.5.2 NTS

For corrective actions identified, commitments are tracked via the Nuclear Tracking System (NTS) which allows for a dependable tracking, searching, and followup system.

4.6 Processes Which Determine The Effectiveness Of Corrective Actions

4.6.1 Root Cause Determinations

Station procedures require that root cause analyses be performed for significant conditions adverse to quality. 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 in accordance with station procedures. The review normally is conducted within twelve (12) months of the implementation of the corrective action. Appropriate actions are taken if the corrective action previously implemented is found to be ineffective.

4.7 Processes For Reporting Problems To The NRC

Issues that become PIFs are required by procedure to be reviewed for reportability. Guidance is provided in the ComEd "Reportability Manual." This controlled manual provides an event driven system of decision trees to aid in reportability determinations and addresses notifications and reporting in accordance with 10 CFR 50.72, 50.73, 50.9, and Part 21, as well as other regulations including 10 CFR 26, 10 CFR 73, 10 CFR 55. The Summary Tables contained in the Manual provide a concise encapsulation of the various reportability requirements.

4.8 Process Effectiveness

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 Process (IRP) occurred relatively recently. However, in general, equivalent processes have been in place throughout the plant's history. Audits and assessments of these processes have been conducted by ComEd personnel and by external agencies, including the NRC. Opportunities for improvement have been identified, and several improvement initiatives have been undertaken. For example, ComEd has engaged a six station peer group to develop a more common, improved Corrective Action Process; and the Nuclear Operations Division Action Plan includes several items intended to enhance the SQV organization, including enhancing the stature of the organizations on site and reviewing SQV staffing levels and competencies.

In March, 1996, the Braidwood SQV organization issued a Level 1 CAR related to the Braidwood Corrective Action Program. This action resulted from a review which identified several examples in which investigations and corrective actions were not completed in a timely manner. The review also identified a case in which a corrective action commitment appeared to have been closed prematurely

The deficiencies were not of the number or nature which would call into question Braidwood's ability to find significant nonconformances with the design bases. However, extensive corrective actions were implemented, which included the following:

- Root Cause Team staffing was increased
- Senior Station Managers are now identified as the responsible persons for all investigations into Level III and above events
- Corrective actions from all investigation events require establishment of completion dates which are tracked to completion by NTS.
- Effectiveness reviews of actions from Level III and above investigations are now being performed
- Procedures and dates for instruction were developed and implemented to clarify expectations for investigation and identification and resolution of trends
- The Station Manager reviews the status of any overdue corrective action items with the responsible Senior Manager and must approve any extensions of completion dates

These actions have been effective in improving the timeliness of completion of investigations and the resultant corrective actions. The Level I CAR was closed in December, 1996. Continued implementation of effectiveness reviews and trending will monitor corrective action program performance on an ongoing basis

A review of Braidwood'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. For example, approximately 75 such items were identified in 1995 and 1996 out of a total of approximately 4000 PIFs. However few (if any) items were significant per the criteria of NUMARC 90-12

1. Does the difference appear to adversely impact a system or component explicitly listed in the TS?
2. Does the item appear to compromise the capability of a system or component to perform as described in the UFSAR?
3. Does the difference appear to adversely impact any applicable licensing commitments?

Two (2) of the 75 items were determined to be reportable as Licensee Event Reports (LERs)

This performance is believed to be indicative of an appropriately high sensitivity to issues related to design bases conformance. Although in some cases the numbers and age of open corrective actions are greater than desired, appropriate resources are being applied to achieve closure in a time frame commensurate with safety significance.

4.9 Summary

Based on the above, 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 any significant nonconformances with the design bases

5.0 Action (e)

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

5.1 Introduction

In summary, ComEd believes that the processes and programs described in this response are effective overall in providing confidence that Braidwood Station is operated and maintained consistent with the design bases and that deviations are reconciled in a timely manner. As detailed in the response to Action (a), the station has a complete set of processes and programs which are designed and implemented consistent with 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 identifies deficiencies, determines their root causes and the extent of occurrence, implements corrective actions, and determines the effectiveness of those corrective actions. 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.

The bases relied upon in reaching this conclusion involve the following elements: (1) consistency with design bases at the time of licensing, (2) controls in programs and processes which 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) ongoing verifications of consistency with design bases as part of normal plant activities; (5) verification of consistency with design bases through self-assessments, NRC inspections and third-party reviews; (6) processes for identifying discrepancies and implementing corrective actions, and (7) continuation of activities that assure ongoing consistency with design bases. Each of these elements is discussed in more detail below.

5.2 Braidwood's Configuration And Performance Were Verified Consistent With The Design Bases As Part Of The Original Licensing Process.

As described in the response to Action (c), ComEd certified that Braidwood had been designed, constructed, and preoperationally tested in a way that would assure consistency with the FSAR, the NRC's 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 Design And Configuration Control Processes Were Established To Control Changes.

As described in response to Action (a), throughout Braidwood's operating history, ComEd has utilized design and configuration control processes which implement the commission's regulations (10 CFR 50 Appendix B, 10 CFR 50.59, 10 CFR 50.71(e), etc.) In general, reviews of these processes by ComEd quality assurance, by the responsible line organizations, and by the NRC, have not identified any significant or programmatic deficiencies in the process procedures, their implementation, or the products. Processes have been improved as necessary to meet heightened expectations and to address weaknesses which were identified. Examples include (1) changes in the use of procedures for valve and equipment lineups and in the Out of Service process, to address the problem of operational configuration control; (2) changes to the UFSAR update process to address UFSAR nonconformances; and (3) changes which require additional management reviews of the progress of design changes, to address the problem of untimely installation and testing. These examples are discussed in more detail in the response to Action (c)

5.4 Braidwood Is Subject To An Ongoing Series Of Verifications As Part Of Normal Plant Operating And Maintenance Activities.

As described in the response to Action (c), Braidwood 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 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 configuration and performance have been maintained consistent with the design bases.

5.5 Braidwood Has Undergone An Extensive Series Of Audits And Inspections.

An extensive series of audits and inspections have been performed which have examined aspects of the plant's conformance with the design bases. In addition to programmatic reviews of the design and configuration control processes (as described in 5.2 above), detailed examinations have been performed of procedures and of the physical plant itself for consistency with the design bases. As described in the responses to Actions (b) and (c), these have included several "vertical-slice" type inspections of key plant systems by the ComEd quality assurance organization, and an NRC EDSFI. Those inspections found generally good consistency among procedures, the physical plant, and the design bases. More significantly, those inspections confirmed that such key plant systems as emergency AC and DC power, auxiliary feedwater, and control room ventilation, remained capable of performing their design bases function.

5.6 Improvement Initiatives Have Been Undertaken Which Have Provided Additional Confidence In Specific Aspects Of The Plant's Conformance With The Design Bases, And Which Have Enhanced The Ability To Maintain Conformance On An Ongoing Basis.

As described in the response to Action (c), Braidwood has undertaken several improvement initiatives which have included one or more of the following types of activities:

- acquiring original design information and improving its accessibility
- revising or establishing more 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

Examples include the acquisition and indexing of the design calculations performed by the NSSS supplier and the architect engineer; the verification or re-performing of calculations for reactor protection and engineered safety features actuation setpoints; and the implementation of a comprehensive MOV program (involving re-performing design calculations, adjusting setpoints, and revising operating and maintenance practices as necessary) to ensure that equipment will reliably perform under design bases conditions.

5.7 Processes Are In Place For Identifying Problems And Implementing Corrective Actions.

As described in the response to Action (d), processes are in place at Braidwood Station which ensure that deviations from the design bases are identified and corrected, and which include appropriate consideration of the need for reporting to the NRC. A review of Braidwood's recent data regarding the identification of problems (PIFs, CARs, LERs, etc.) indicates that the processes have been successful in identifying and correcting problems related to design bases conformance. The fact that a number of items of low significance have been identified is believed to be indicative of an appropriately high sensitivity to issues related to design bases conformance. ComEd management identifies and acts upon adverse trends. An example is management's response to the results of the review of selected UFSAR actions. As described in the response to Action (c), ComEd formed a Process Improvement Team to develop an improved UFSAR update process for all Stations, and all stations will review additional UFSAR actions. (The extent, schedule, and relationship of these reviews to other ongoing initiatives is presently under evaluation.)

5.8 Design Bases Conformance Activities Will Continue

The 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, SQV 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. Taken together, the assurance already established by these programs and processes and the management's continuing commitment to improvement demonstrate that the station will continue to apply and improve its ability to maintain the plant consistent with its design bases.

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

1.0 Role of ComEd Engineering in Design Bases Management

The Site Engineering Organization plays a significant role in controlling, maintaining, and assuring 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 t 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 are described below

1.1 Transition of Design Control and Engineering In-House Development

ComEd's historical approach to design had been the use of a Corporate Engineering Department producing design and analysis predominantly by managing architect engineering (AE) contracts from the General Office ComEd utilized an AE Guidebook which formalized the interfaces and communication channels between ComEd and the AE. At Braidwood the AE that performed the original design was utilized as the primary AE for design activities after receiving the operating license.

Problem solving and system engineering functions were organized under a technical staff that reported on site to the Station Manager. The responsibility for design of the reactor core was centralized at the General Office, initially utilizing the NSSS suppliers for the design. A transition was begun in 1990 for core design to be performed in-house

In late 1992, the nuclear organization was changed to establish design engineering authority and accountability on-site under a Site Vice President.

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 AE's 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 in its own direction for understanding and control of the design bases The TENERA self-assessment identified eleven strategic issues and targeted recommendations to deal with those issues. Key among them was the understanding and "owning" of the design. The self-assessment recommended that ComEd become more knowledgeable in the design, license, and operating bases of the plants; and that we be in a stronger position to control the design configuration and be proactive in any matter which requires design information 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. An Engineering Vice President position was established. ComEd established a vision which assigned to the engineering organization the primary responsibility to be accountable to prevent and solve problems.

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 site's existing accredited training program to ensure that the engineers onsite have a common foundation in engineering fundamentals, plant systems, and site processes.

1.2 Relative Roles of Corporate and Site Engineering

As indicated above, the corporate office evolved from being the principal 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, PRA, and common multisite projects, e.g., power uprate and 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. The key information transfer vehicles that have been used are a daily engineering phone call, a Tech Alert program, Corporate Engine ng oversight of station activities, the Engineering Managers Team meeting, and Engineering Peer Groups.

Tech Alerts - Tech Alerts are prepared and issued by Corporate Engineering 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.

Corporate Engineering Oversight Role - The Chief's staffs review selected design products developed by the Site Engineering organizations. The objective of the reviews is to assure that the design is adequate and is in compliance with all procedures.

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

The departments at our stations share a responsibility in maintaining Configuration Management. Engineering is accountable for ensuring that the physical plant is in conformance with the design bases; 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 maintenance work is 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 commonly across the six nuclear stations.

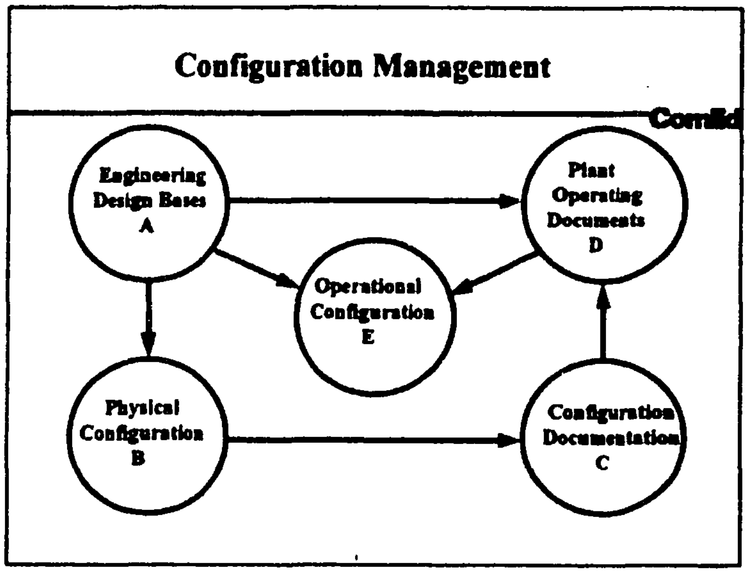
At each of the six nuclear stations, there is a supervisor in the site Engineering Department who is accountable for configuration control. This supervisor oversees the design change processes discussed in Action (a), and oversees the close-out of the design changes to ensure all controlled documentation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

1.4 Configuration Management Model

The following "five ball" model illustrates ComEd's approach to Configuration Management:

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).

Acti n (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 Assessment Organizations and Departments

ComEd implements many 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.

This section summarizes the role of Corporate Engineering Assurance, Offsite Reviews, and Quality Verification Services. It also discusses a new function established in January 1997 at all six stations: Site Engineering Assurance. Some of these roles, in particular the Site Engineering Assurance role, were established to provide an extra level of reviews of products prepared by the Site Engineering Organizations.

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 a.. .rance 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 will lead a peer group of the newly established 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 site engineering activities, onsite Engineering Assurance organizations directly reporting to the Site Engineering Manager have been established.

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

1. Design Change Activities
2. Operability Evaluations
3. Safety Evaluations
4. Calculations
5. Surveillance Trending
6. Special Test Procedures

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

1. Verify that the activity is enveloped by the Station's licensing and design bases.
2. Challenge the basis and adequacy of Safety and Operability Evaluations.
3. Assess whether calculations use conservative methodology and assumptions.

This activity is not a substitute for any reviews currently implemented in the existing design control processes. It is intended to be near real-time with respect to the engineering activity being evaluated. The oversight function will foster a questioning attitude with regards to the licensing and design bases of the station.

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 Safety Evaluations and Licensee 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 Onsite Review/Plant Operations Review Committee meetings.

2.4 Corporate Quality Assurance and Site Quality Verification (SQV)

The Nuclear Oversight Manager manages the Quality Assurance Program 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 line and Corporate activities to ensure compliance with quality assurance and nuclear safety requirements. This position reports to the Site Vice President. He monitors the day-to-day station activities involving operating, modification, maintenance, in-service inspection, refueling and stores through onsite audits, field monitoring, and safety reviews.

Matrix of Appendix II Processes

Process Number	Process Description	Procedure Reference	Implements Regulatory Requirement		Configuration Management Model Linkages (note 1)						
			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 NEP-04-02	X	X	X	X	X	X	X		
5	Modification Work Control Process	NSWP-G-01 (note 2)	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	Setpoint Change Request	Station Procedure	X	X	X	X	X	X			
11	Engineering Software Development and Revision Process	NEP-20-01	X		X						
12	Engineering Change Notices (ECNs)	NEP-08-01				X	X	X			
13	Safety Evaluation Process	NSWP-A-04	X	X	X						
14	VETIP Processing	NEP-07-04				X	X	X			
15	Configuration Control Using EWCS	NEP-14-01				X	X				
17	Calculation Process	NEP-12-02			X	X					
18	Operability Determination Process	Station Procedure (note 3)			X	X			X		
19	UFSAR Update Process	Station Procedure (note 3)	X	X	X	X	X	X	X		
20	Out of Service/Return to Service Process	Station Procedure (note 3)								X	X

NOTES:
1. A/A Link are processes that affect the design bases only.
2. Applies to Field Change Request when needed.
3. Since Processes 18-20 are station unique they are discussed in Action (a).

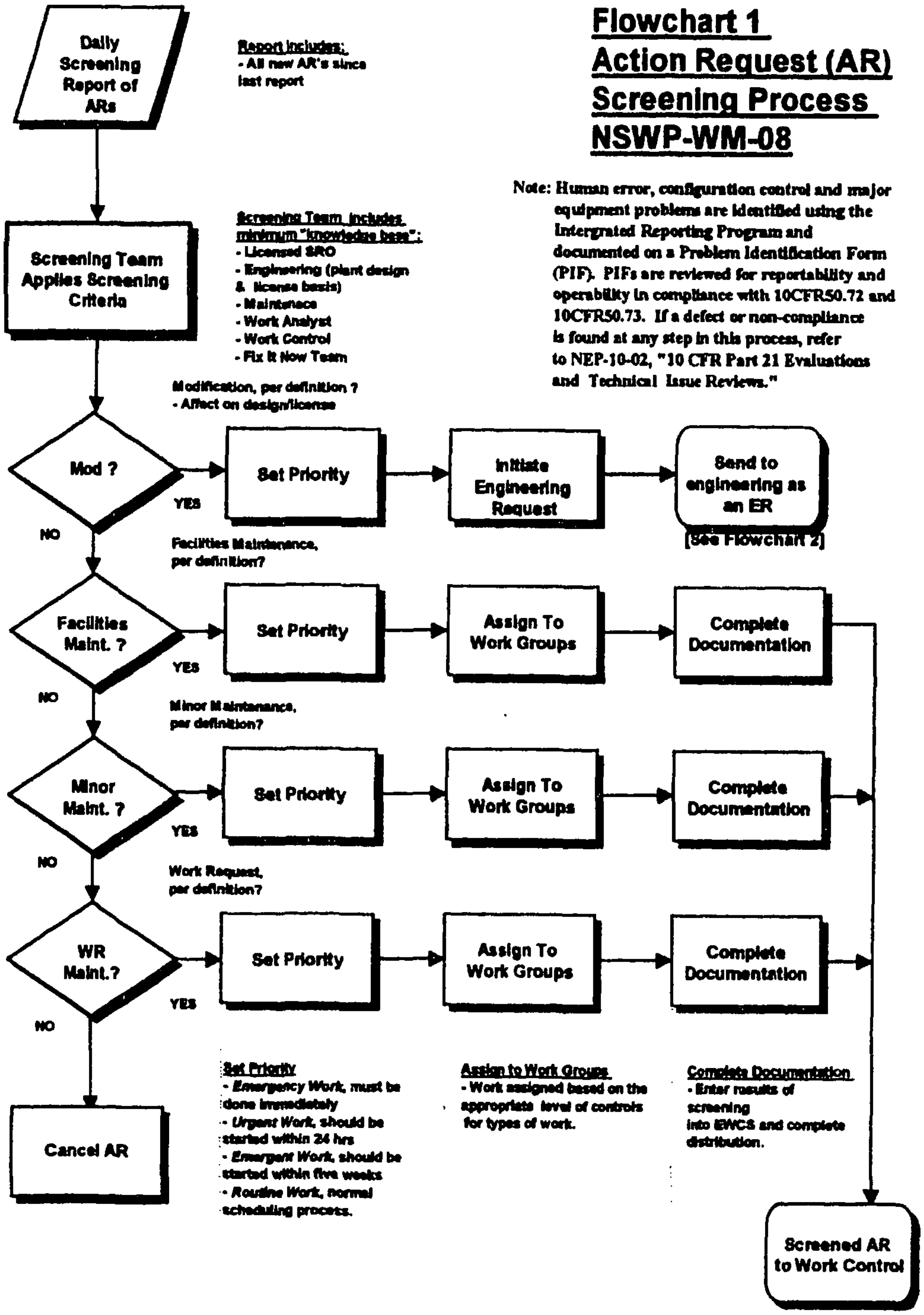
Appendix II Design Control and Configuration Control Processes

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 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 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. 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."



Action Request (AR) Screening Process

NSWP-WM-08

PURPOSE

Work that needs to be done at ComEd's nuclear stations, is initially identified and documented on an Action Request (AR) 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 basis.
- 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" requirements 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 basis understanding from people who often reference and interpret the appropriate source documents. If the person representing Engineering is unfamiliar with the proposed work and its affect on the design/licensing basis, they will know who to contact.

The second line of defense in ensuring that work is performed with 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 basis of the plant and to ensure that no unrecognized design changes are being made.

Additionally, with recent industry and ComEd events (especially the LaSalle Service Water event) that deal with design/licensing basis issues, an increased awareness of the affects 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 affect on the design basis of the plant. This was formalized with the recent issue of NSWP-WM-08, Action Request Screening.

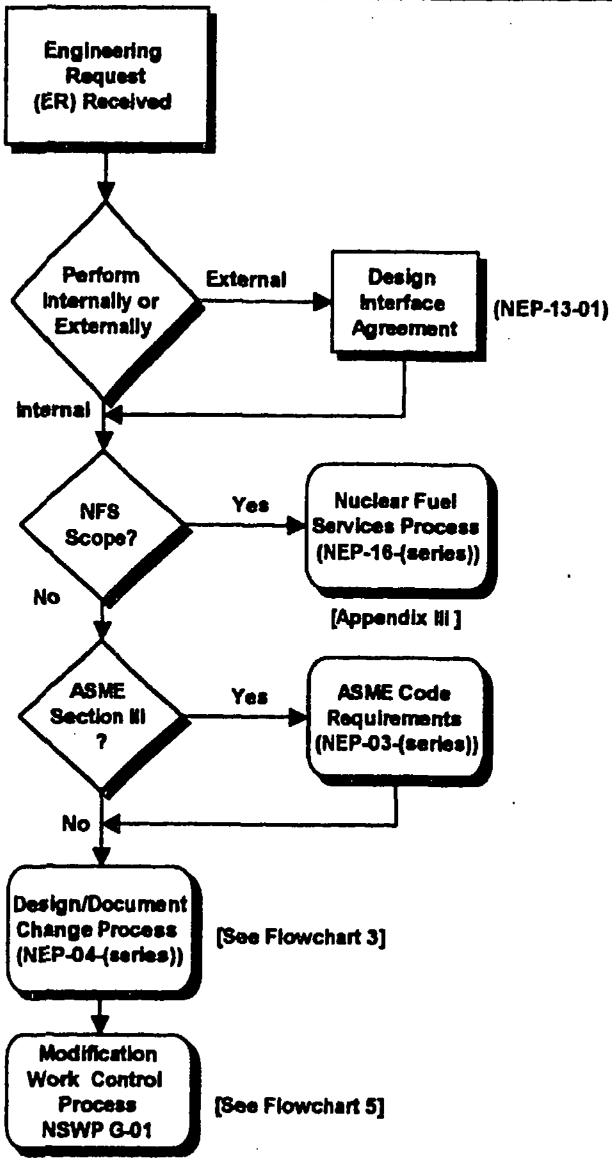
Increased emphasis has also been placed on the definition of Facility Maintenance, Minor Maintenance, and Work Request Maintenance. In each of these types of work, clear boundaries have been provided to maintain the appropriate level of controls. If during the process something requires work to fall 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. 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 any 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.

With the introduction of EWCS, the methodology has changed somewhat but the intent of the process is unchanged. Decisions on safety classification will only be required on an exception basis when the classification of components has been captured in the data base supporting the process. Additionally, organization changes have taken place with the creation of Work Control Centers and the screening function was typically reassigned to the Lead Unit Planners and Lead Maintenance Planners. While this has worked well in most cases, inadequate sensitivity to Action Requests with the potential to introduce changes to the design has occasionally been observed. Further, Minor Maintenance teams and Fix It Now teams have also been created which have predefined boundaries in which they perform specific types of work. In response, 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 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 Reviews."

Roadmap to Design Control Process

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 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."

RECENT/PLANNED IMPROVEMENTS

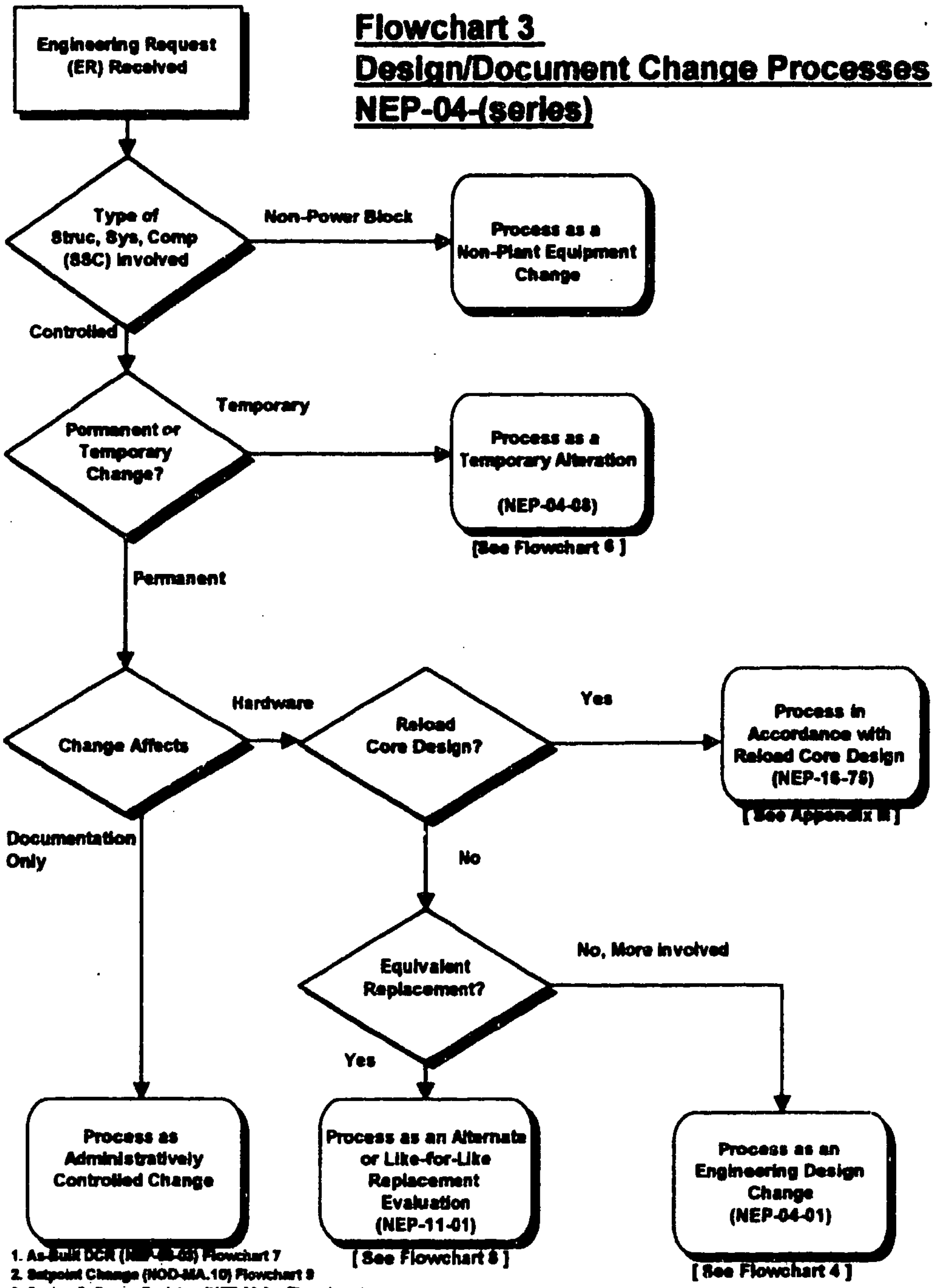
Improvements to the processes represented here are discussed in the detailed process descriptions that follow. As stated in the November 12, 1996 letter from T. Maiman to A. Bill Beach, ComEd will establish an Engineering Assurance Group to provide oversight of key engineering activities. This organization will remain in place until normal engineering activities have improved to the point where these reviews are no longer required.

As stated in the November 12, 1996 letter from T.J. Maiman to A. Bill Beach, ComEd will expand the SQV audits of our major design contractors with focus on:

(1) Interfaces with ComEd,
(2) Design control processes, and
(3) Corrective action notification

An action plan will be developed that includes the principal architect-engineers, fuel suppliers, and NSSS vendors.

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



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

Design/Document Change Processes

NEP-04-(Series)

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-Plant Equipment Changes - The first decision point determination is whether the proposed change can be processed as a Non-Plant Equipment 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 any Controlled operable SSC, 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 the 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

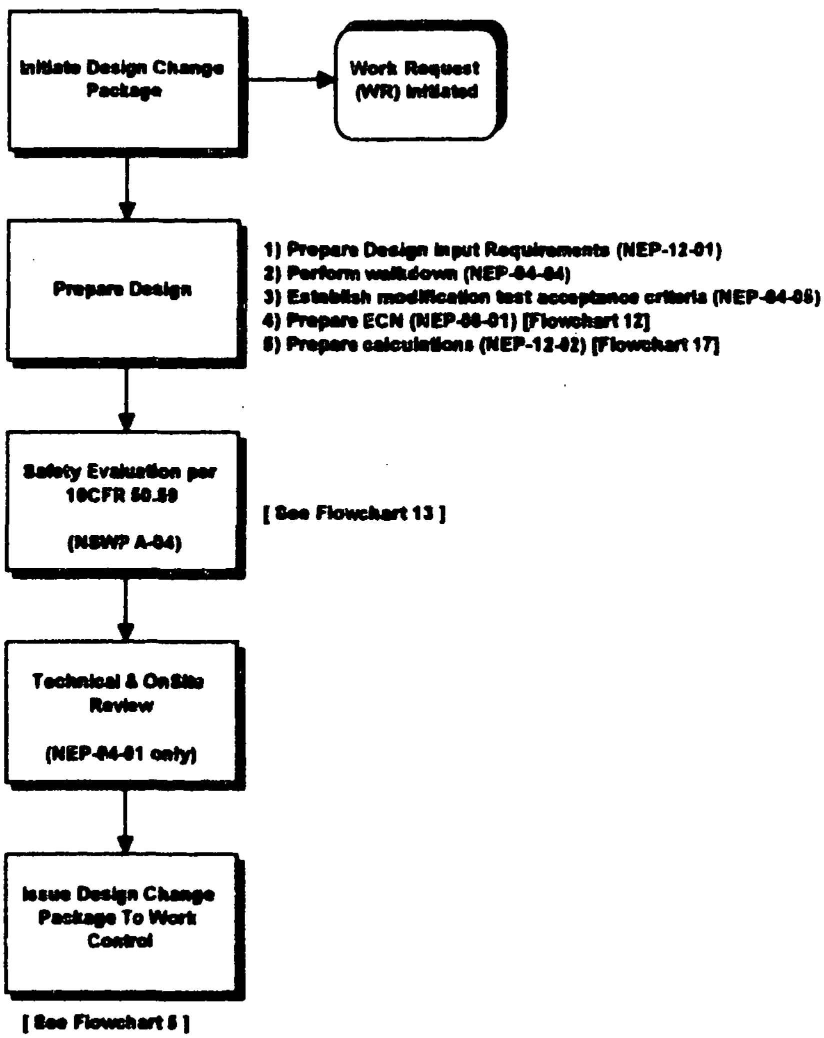
CHECKS AND BALANCES

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, Braidwood has established a separate process for handling these "Non-Plant Equipment Changes." 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 4
Engineering Design Change Process
NEP-04-01 and NEP-04-02



Engineering Design Change Process

NEP-04-01 and NEP-04-02

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 basis 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 basis,

- 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 may be held (optional for Exempt Changes) 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 any potential concerns or problems If the design has a low potential to significantly reduce the margin of nuclear safety and requires minimal engineering input, it is categorized as an "Exempt Change" and is processed in accordance with NEP-04-02 If the ER is approved as a Controlled Design Change, it is processed in accordance with NEP-04-01 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 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 This final review is not required for Exempt Changes.

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 basis. The first area is handled through Engineering Change Notices (ECNs), which are used to develop the detailed design for safety related Design Changes. 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 50.59 safety evaluation is also part of the design process and provides an additional level of review. The ECN, calculation and safety evaluation process are described separately in more detail.

When the design is installed "out 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 50.59 safety evaluations may be required. 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

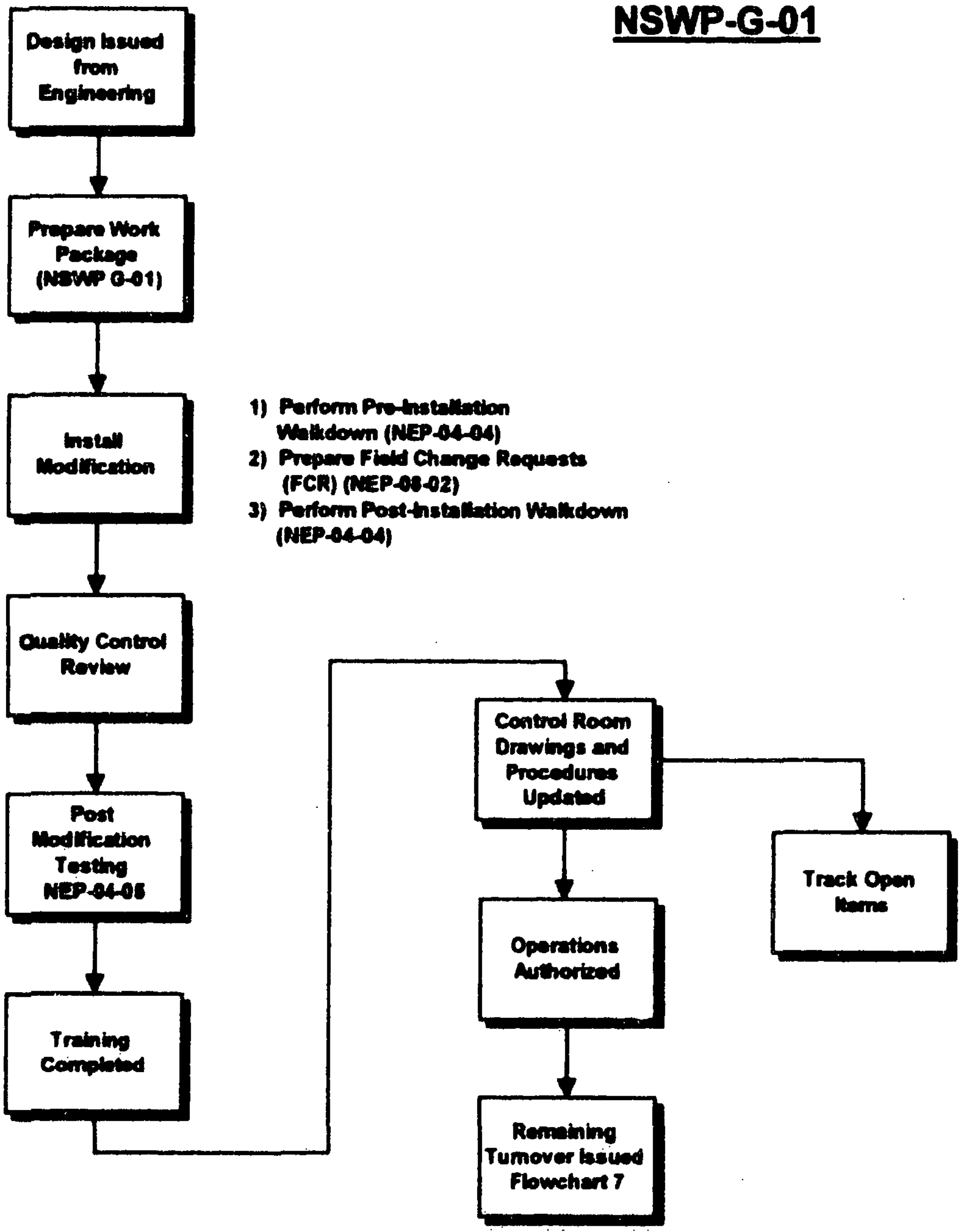
Post Modification Testing, as discussed in the Modification Work Control Process, is the last area where the design is evaluated to ensure that it has met the original design requirements.

RECENT/PLANNED IMPROVEMENTS

Improvements in the ECN, calculation and safety evaluation processes are addressed in the specific process descriptions

As stated in the November 12, 1996 letter from T.J. Maiman to A. Bill Beach, ComEd recently completed a review of partially implemented modifications, and established a schedule to close them out in a timely manner.

Flowchart 5 Modification Work Control Process NSWP-G-01



Modification Work Control Process

NSWP-G-01

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 approved, a Work Package is prepared that provides the necessary instructions for installation, 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 Any 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

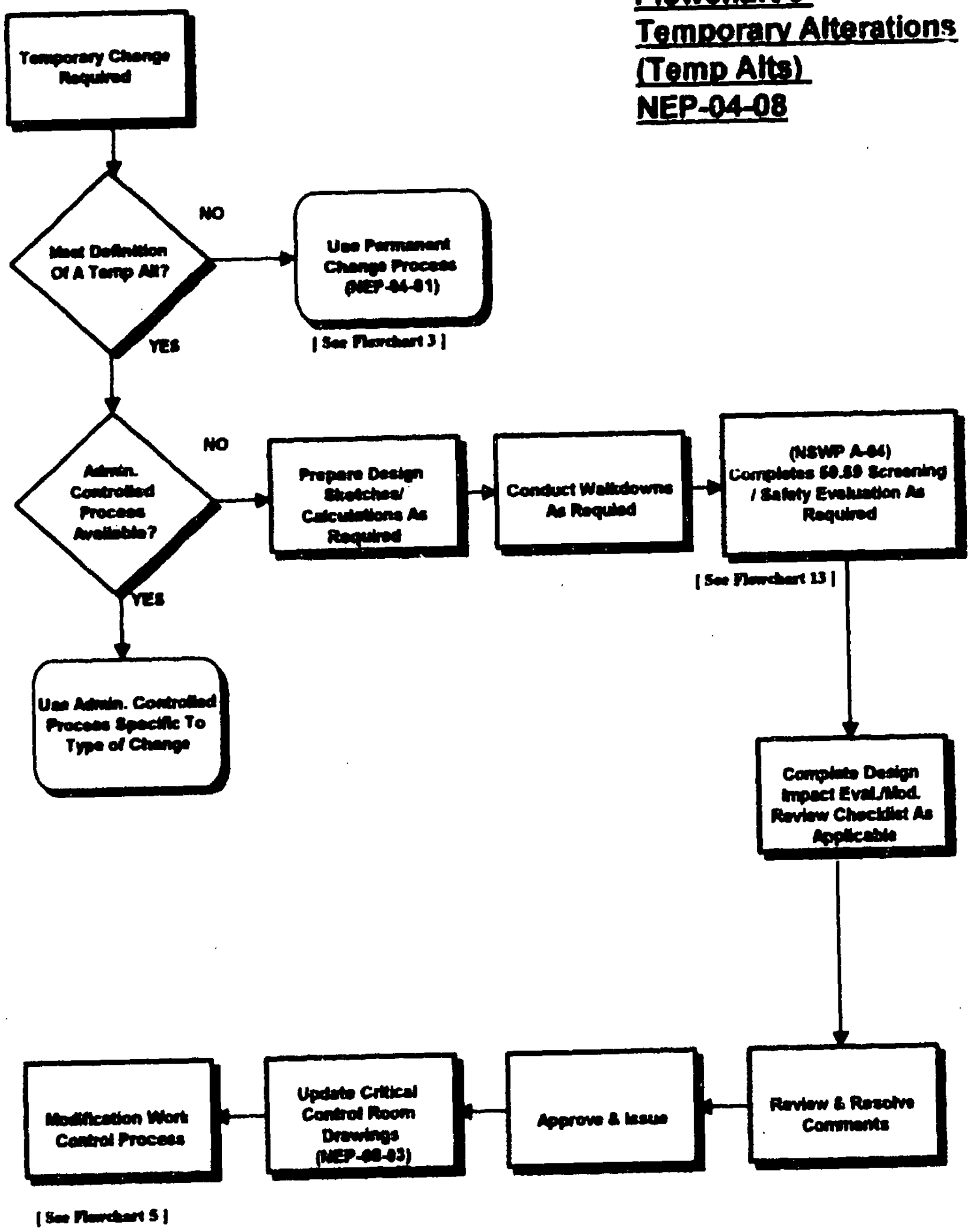
The pre-installation walkdowns provide an opportunity to evaluate the modification against the physical attributes and design considerations of other components located in same area Any changes required during this evaluation and others required during the installation, are all evaluated through the 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 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 6
Temporary Alterations
(Temp Alts)
NEP-04-08



Temporary Alterations (Temp Alts) Process
NEP-04-08

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 50.59 screening/safety evaluation is completed, as appropriate.

A Design Impact Evaluation and Modification Review Checklist are completed, as applicable. 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 50.59 screening/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 consideration are addressed.

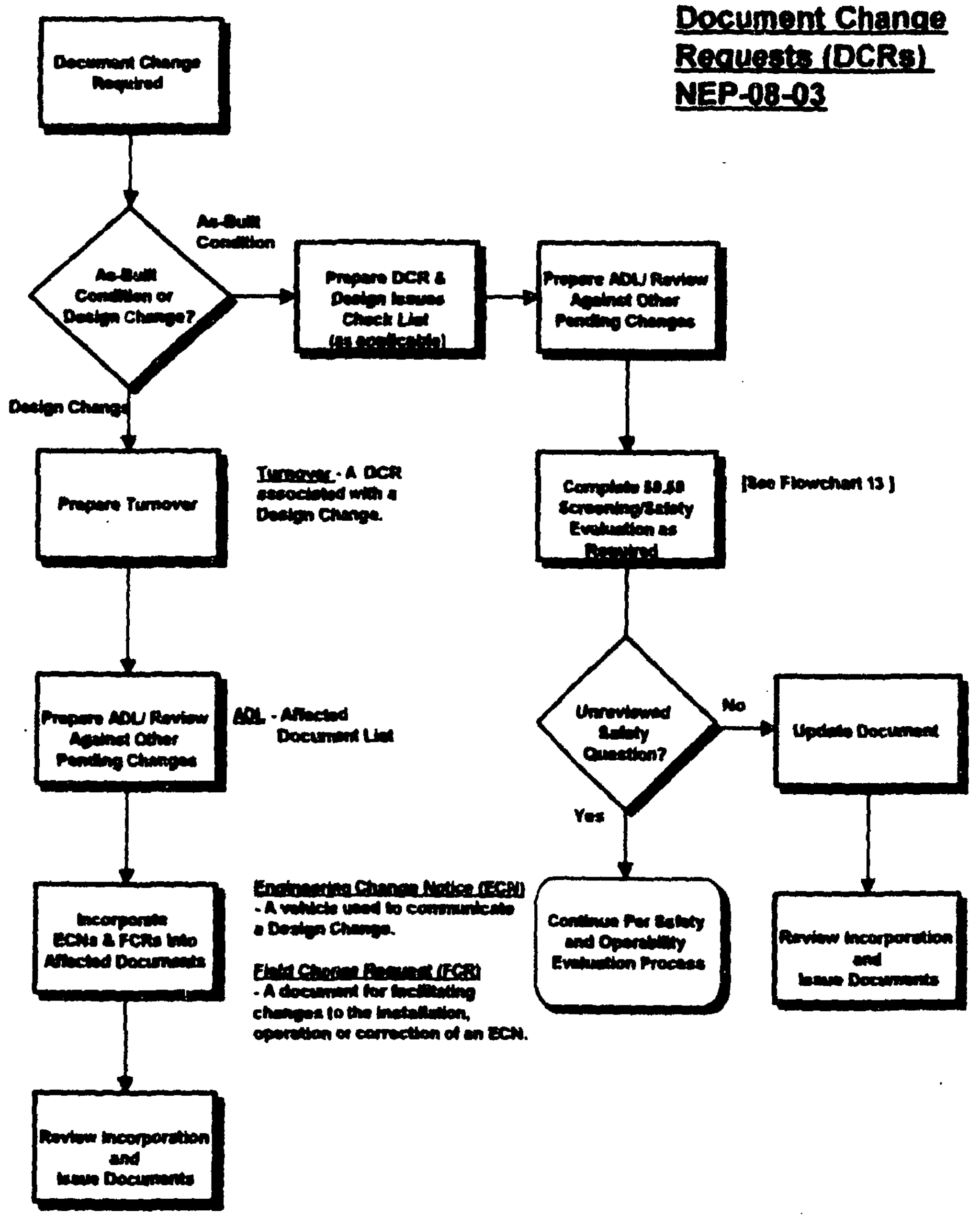
Temporary Alterations are required to be updated on the Critical Control Room Drawings (CCRD) so that these are maintained to reflect the plant configuration at all times.

RECENT/PLANNED IMPROVEMENTS

There is currently a six site evaluation team that has been formed to review Temp Alt issues that were identified through Nuclear Regulatory Commission, Site Quality Verification, and Chief Design Review. This team has established root causes and solutions that are now being reflected in a new NSWP.

This new NSWP is intended to simplify the process, improve the understanding of what is considered a Temp Alt and standardize the process at all six sites. Implementation is planned for early 1997.

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



Document Change Requests (DCRs)

NEP-08-03

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 50.59 Screening/Safety Evaluation is prepared. If no Unreviewed Safety Question has been identified, the documents are updated, reviewed, approved, and issued.

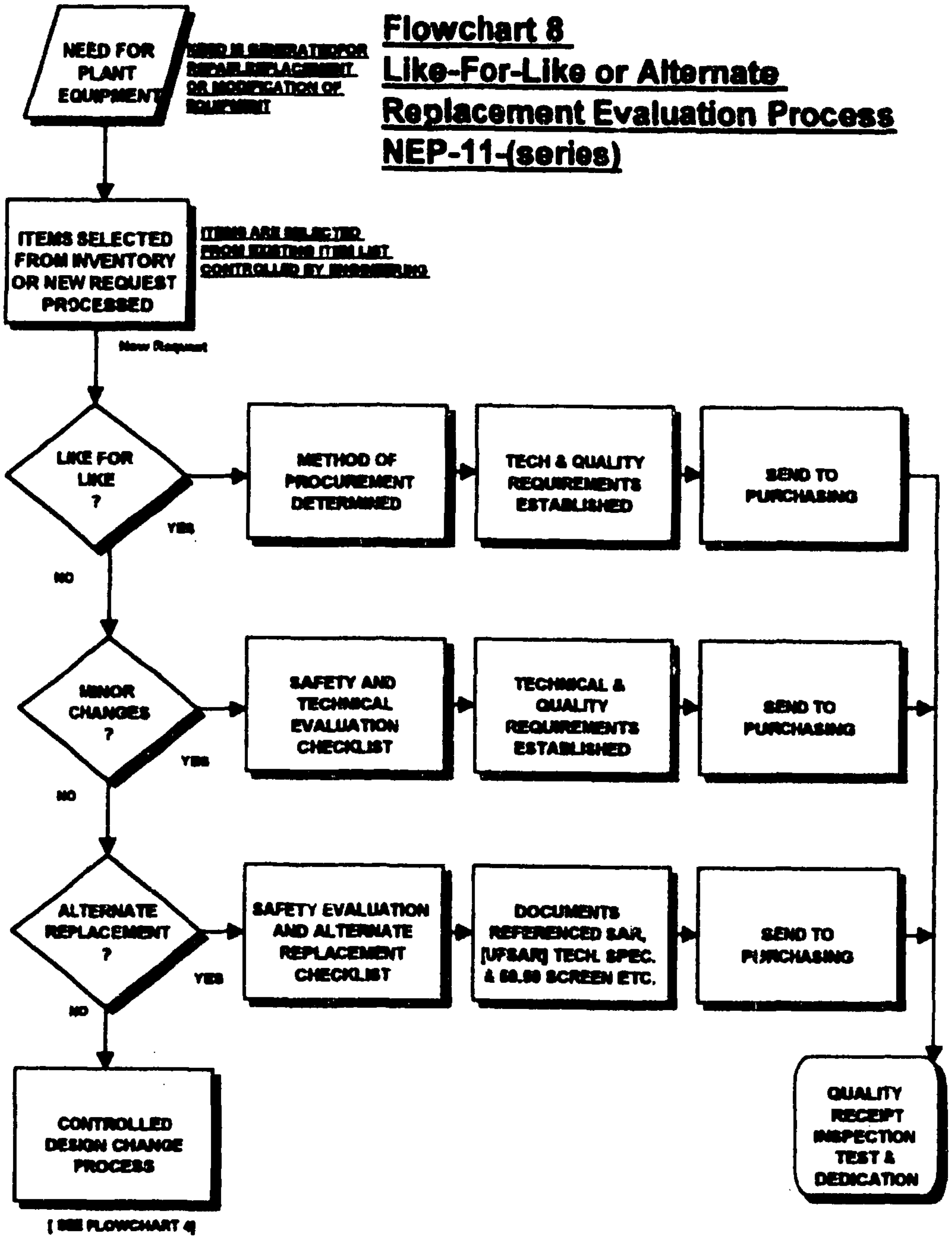
CHECKS AND BALANCES

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



Like-For-Like or Alternate Replacement Evaluation Process

NEP-11-(Series)

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 Basis
- 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.

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 50.59 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:

- 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. 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 Hartford 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 and Weaknesses

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 other military, 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.

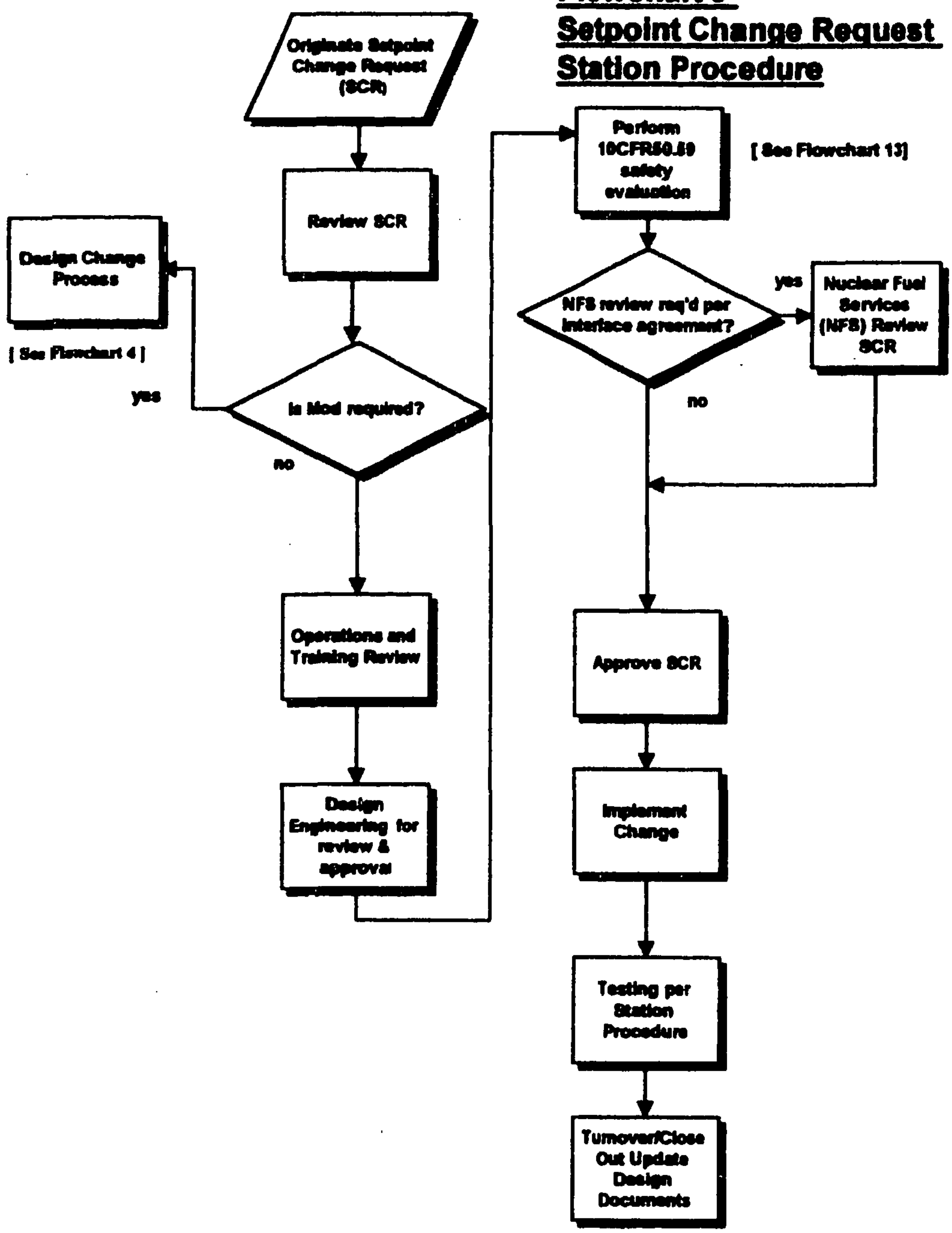
RECENT/PLANNED IMPROVEMENTS

A timeline entitled, "Safety Related Materials Process Improvements" is attached to summarize the issues and corrective actions that have been implemented in the materials procurement process from 1988 through 1996. This provides a ready summary of the improvement efforts undertaken during this time frame.

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 plants 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 INPO ACAD criteria.

Flowchart 9
Setpoint Change Request Station Procedure



Setpoint Change Request Process

Station Procedure

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 mod 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 safety review including a 10 CFR 50.59 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 (if necessary).
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 Turnover shall be initiated to update the appropriate design documents and/or data-bases.

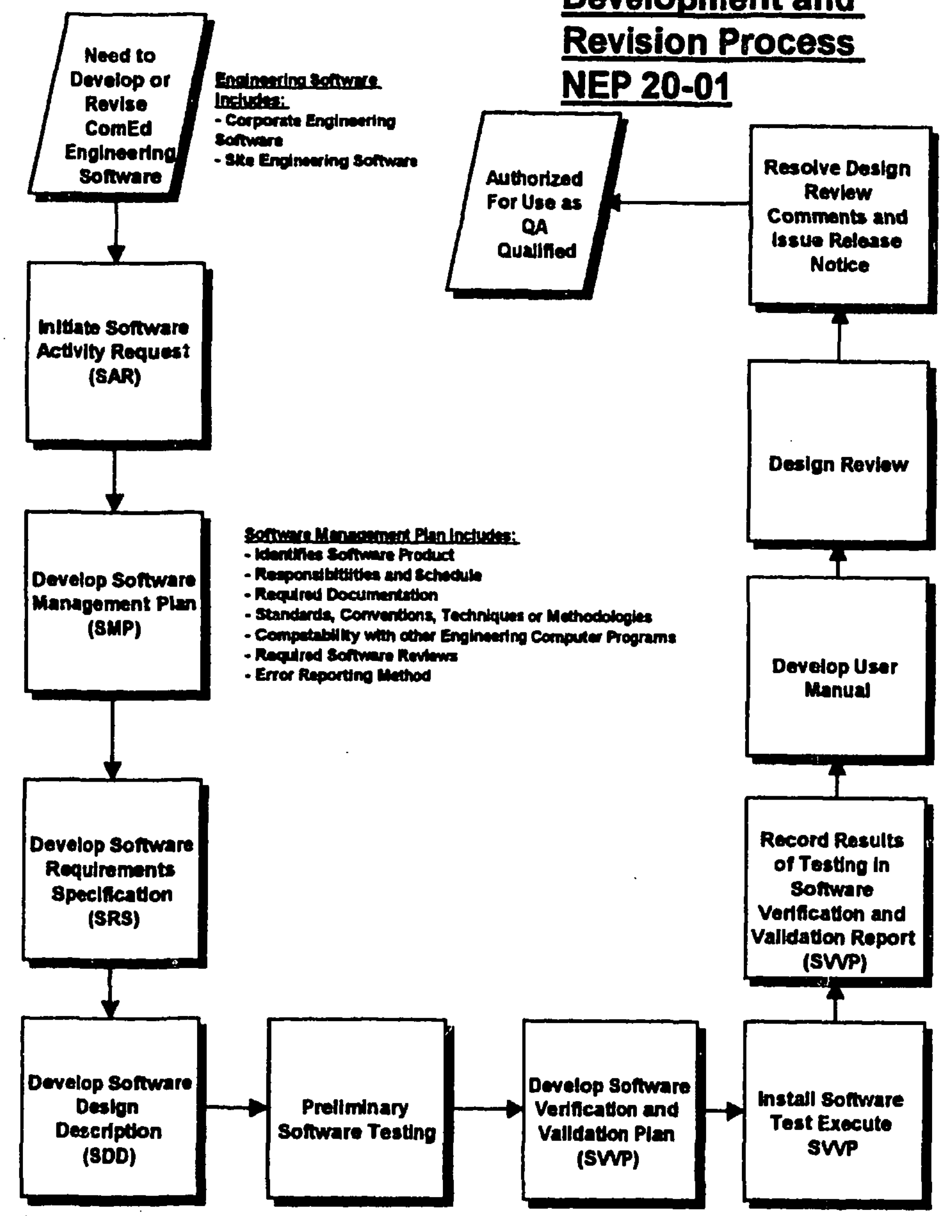
CHECKS AND BALANCES

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 11
Engineering Software Development and Revision Process
NEP 20-01



Engineering Software Development and Revision Process

NEP-20-01

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 ECP to assure that the software produces correct results for the test case.

CHECKS AND BALANCES

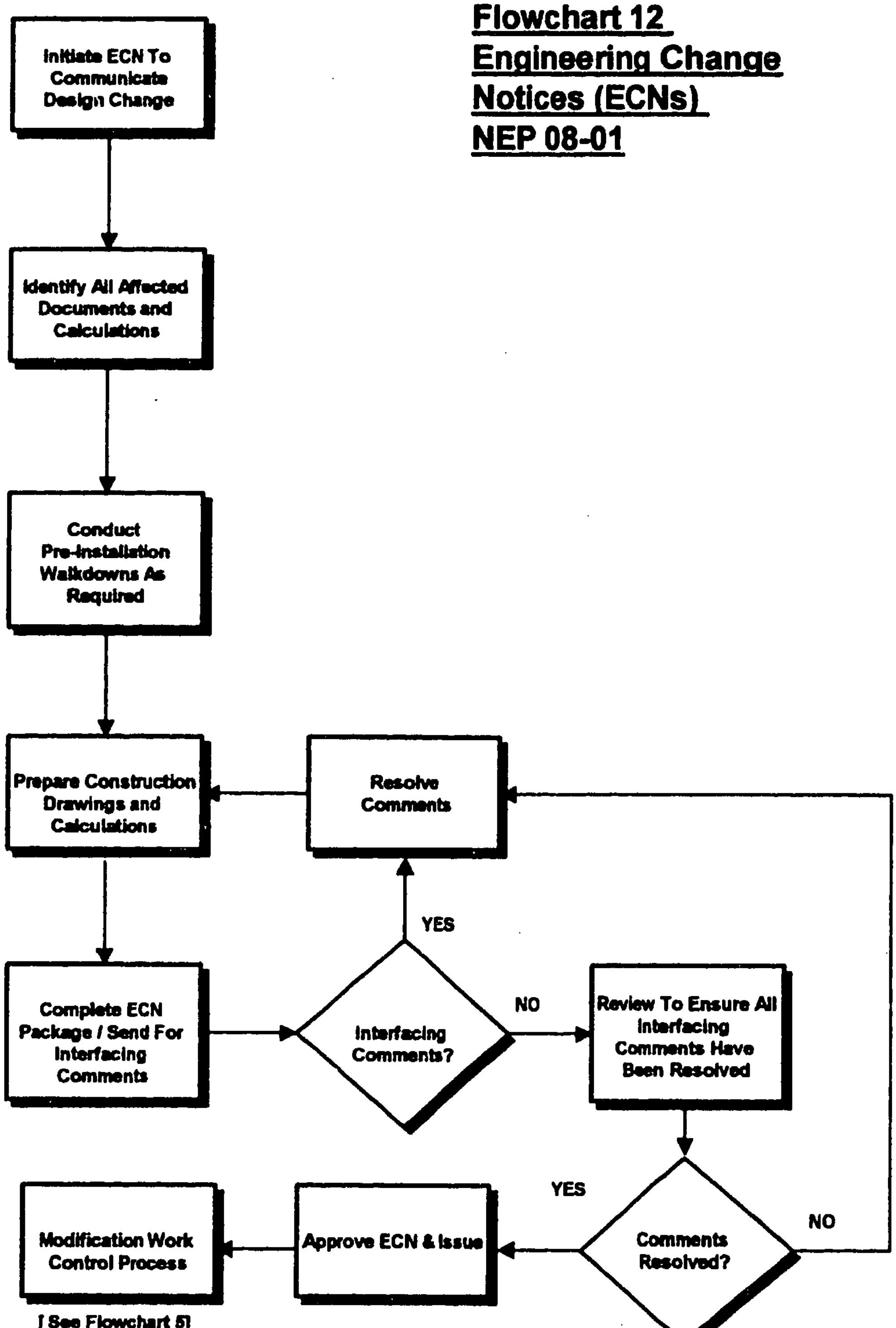
Software Verification and Validation (SVV) activities shall begin with the development of a SVV Plan which shall describe:
- tasks and criteria for accomplishing the Verification of the ECP.
- hardware and software configurations pertinent to V and V.
- tracability to both the software requirements and the software design.

The software shall then be 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 12 Engineering Change Notices (ECNs) NEP 08-01



Engineering Change Notices (ECNs)

NEP-08-01

PURPOSE

Engineering Change Notices (ECNs) are used to communicate design changes which are included in a safety related 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 process.

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 sent 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.

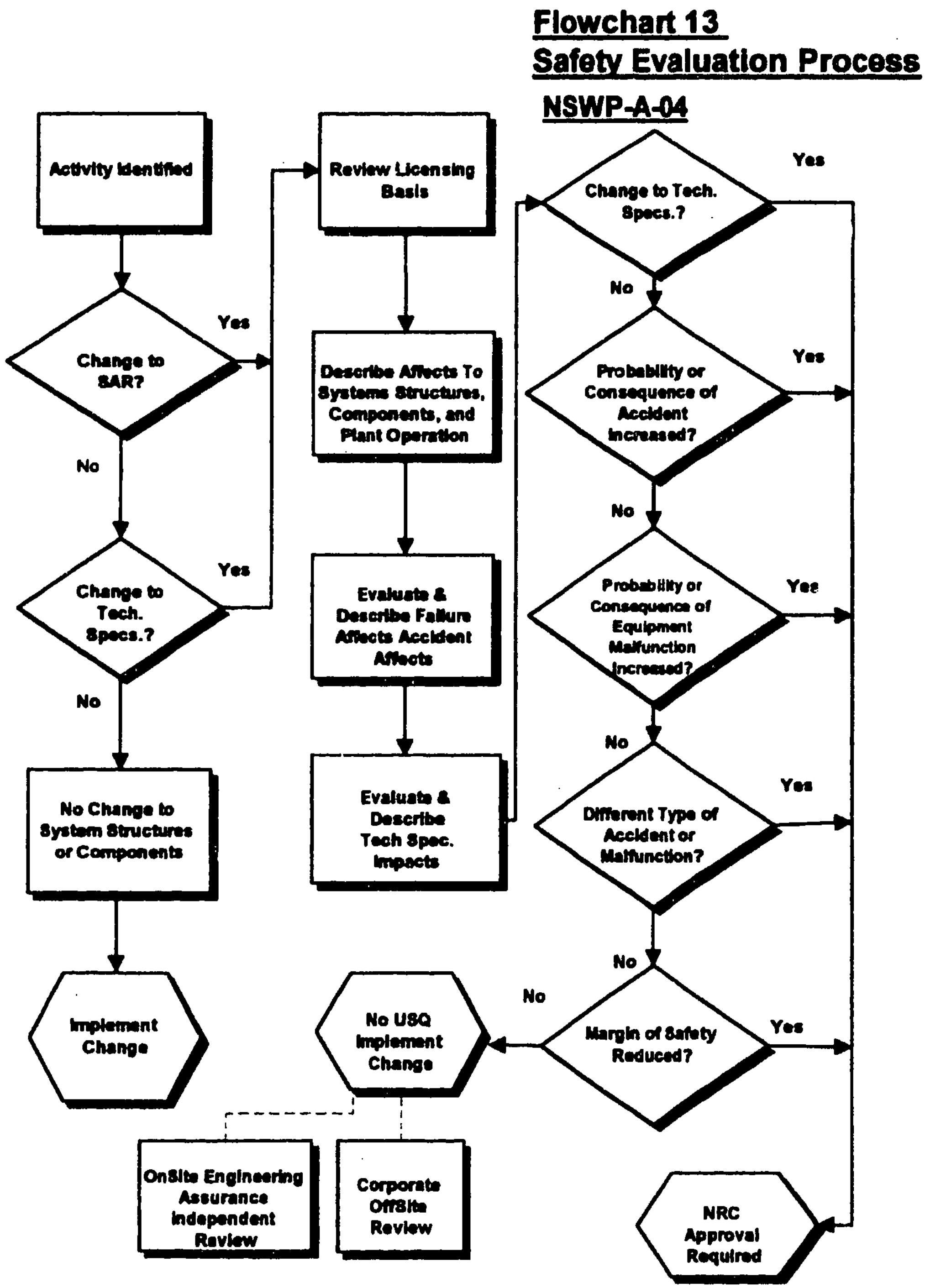
CHECKS AND BALANCES

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 13
Safety Evaluation Process
NSWP-A-04
Activity Identified
Change to SAR?
Yes
No
Change to Tech. Specs.?
Yes
No
No Change to System Structures or Components
Implement Change
Review Licensing Basis
Describe Affects To Systems Structures, Components, and Plant Operation
Evaluate & Describe Failure Affects Accident Affects
Evaluate & Describe Tech Spec. Impacts
Change to Tech. Specs.?
Yes
No
Probability or Consequence of Accident Increased?
Yes
No
Probability or Consequence of Equipment Malfunction Increased?
Yes
No
Different Type of Accident or Malfunction?
Yes
No
Margin of Safety Reduced?
Yes
No
No USQ Implement Change
OnSite Engineering Assurance Independent Review
Corporate OffSite Review
NRC Approval Required

Safety Evaluation Process

NSWP-A-04

PURPOSE

To determine and provide a documented basis for concluding if an Unreviewed Safety Question exists for a change, test, or experiment.

PROCESS DESCRIPTION

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

A Screening is performed to determine if the change can directly or indirectly affect any of the requirements of the UFSAR, Technical Specification, and other licensing basis information.

If any affects are determined a Safety Evaluation must be performed to determine if the change could result in an Unreviewed Safety Question.

The Reviewer reviews the UFSAR, pending UFSAR changes, and other Licensing Basis documents.

Describe how the proposed activity will affect plant operations.

Describe how the proposed activity will affect equipment failures.

Identify accidents / transients activity could affect.

Determine if new or revised Tech Specs are needed.

For each accident affected, discuss the probability of the accident being increased.

For each accident affected, discuss the effect on the consequences of the accident.

Discuss how the activity affects the probability of a malfunction of equipment important to safety.

Discuss how the activity affects the consequences of a malfunction of equipment important to safety.

Discuss the possibility of a new accident or malfunction of a type different than those previously evaluated in the SAR

For each Tech Spec involved with the activity determine affects on acceptance limits and margins.

Determine if the margin of safety is reduced.

Identify if changes to the SAR are needed to complete the activity.

All completed 50.59 reviews are to be independently reviewed by the Off Site Review Group for the following:

- Confirm the conclusion of no USQ
- All questions are properly answered
- Supporting documentation justifies conclusion
- Technical Specification change needed

CHECKS AND BALANCES

The overviews of the safety evaluations performed by the on-site Engineering Assurance group and the Offsite Review Group provide additional levels of independent assessment of the quality and effectiveness of this process.

RECENT/PLANNED IMPROVEMENTS

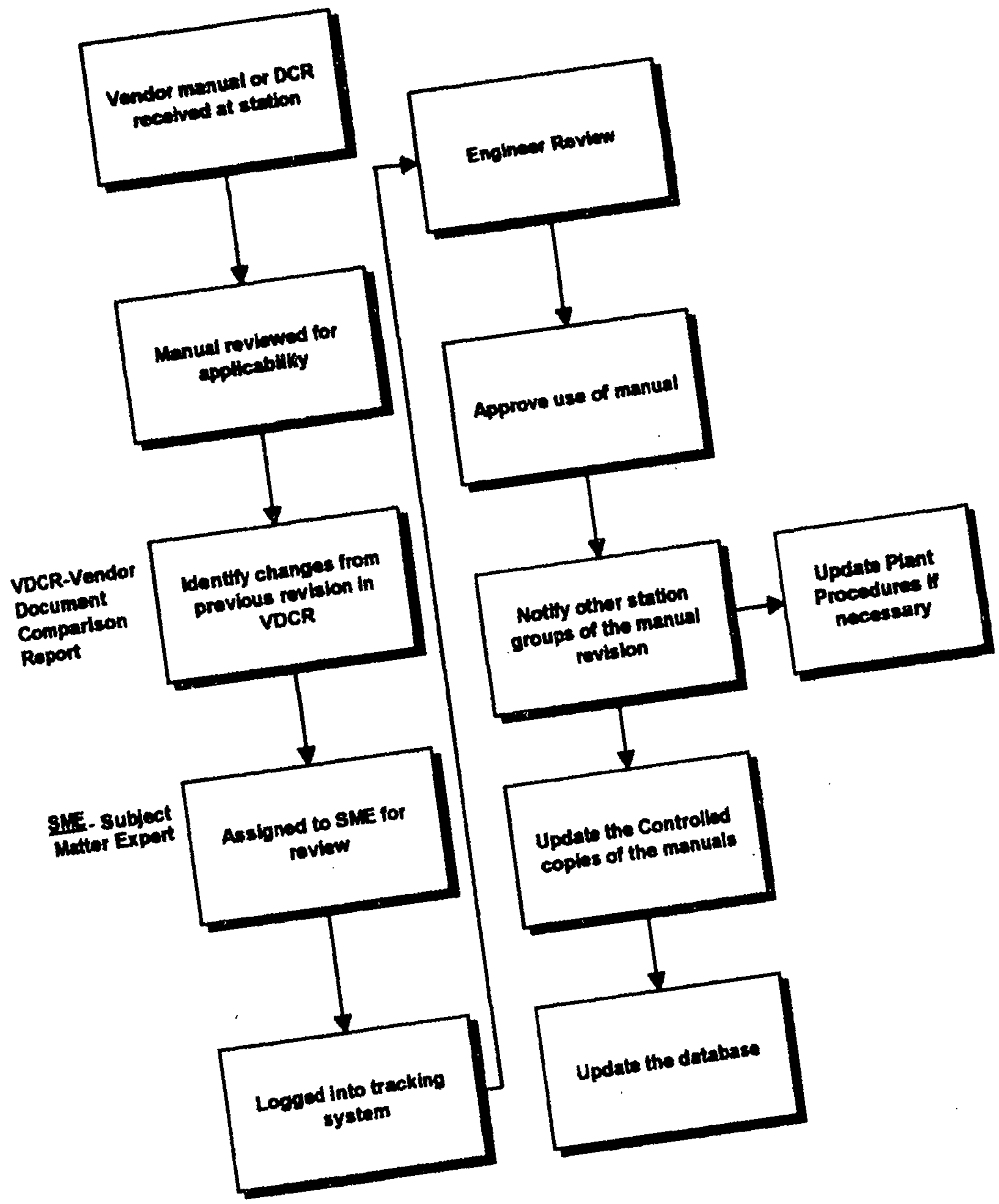
ComEd's Safety Evaluation Process has been the subject of several NRC Audits. The following steps have been implemented on a corporate-wide basis to improve this process:

A common, Corporate procedure has been developed for use by all departments, by all stations. Previously, each station had different procedures and, in some cases, different procedures for different departments. This Corporate procedure is scheduled for implementation in the first quarter of 1997.

A Chief Engineer, in charge of regulatory compliance has been assigned accountability to teach and mentor Corporate and Site Safety Evaluations. Training and certification is required of all Safety Evaluations

As stated in the November 12, 1996 letter from T.J. Maiman to A. Bill Beach, ComEd has established Engineering Assurance teams to review operability and safety evaluations.

Flowchart 14
VETIP Processing
NEP-07-04



VETIP Process

NEP-07-04

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 Letter 90-03, all vendors supplying critical safety related components are recontacted 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:

A review for applicability will be done by the VETIP Coordinator. This step also includes a review to see if the document is already in use at the station.

If the vendor manual is a revision to an existing manual, a review to classify the document as an administrative or technical change is made.

If the vendor manual is a revision to an existing manual, a summary of revisions document, called a Vendor Document Comparison Report (VDCR), is prepared.

Review of the changes to the vendor manual by the Subject Matter Expert (SME).

SME approves the manual, as appropriate, and determines what other station groups should be notified of the manual change for their work. If station procedures are affected, the manual is sent to the procedure coordinator.

VETIP Coordinator processes the new vendor manual and updates hard copies and databases.

CHECKS AND BALANCES

The Subject Matter Expert Review concept is new and ensures the right person is reviewing the manual and no time is lost waiting for other reviews.

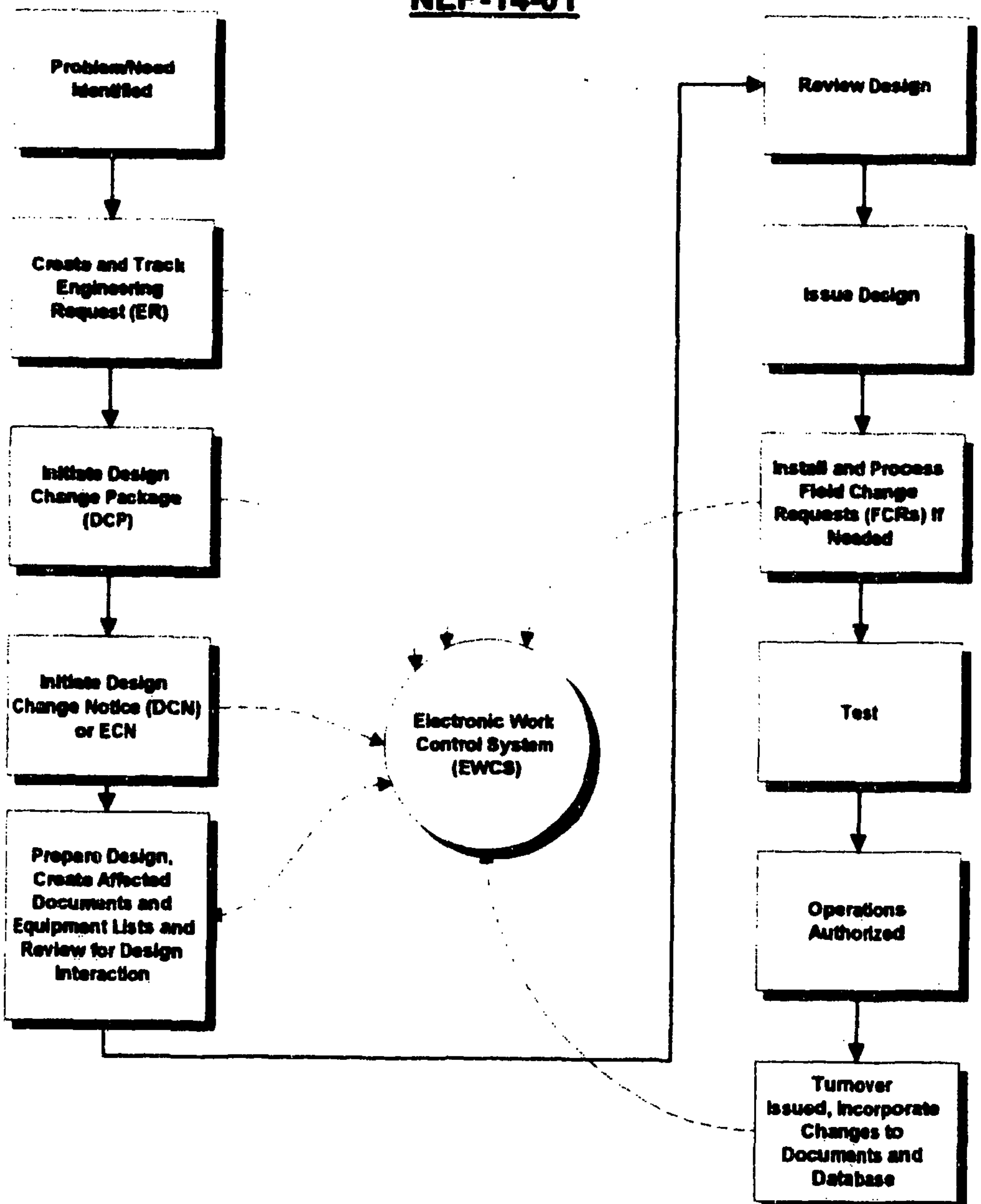
Common processing at each station for better control and a more consistent review and documentation of VETIP information.

RECENT/PLANNED IMPROVEMENTS

Procedural requirements for processing incoming vendor manuals within a 90 day period has not always been met because of emergent work The stations are adding contractors to help eliminate the backlog of old documents and to get the program current with incoming work

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

Flowchart 15
Configuration Control Using EWCS
NEP-14-01



Configuration Control Using EWCS

NEP-14-01

PURPOSE

The Electronic Work Control System (EWCS) is an online 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 (future)
- Controlled Documents (CD)
- Equipment Database (future)

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. (future)

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) (future)

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 (future)

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 any other open change 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.

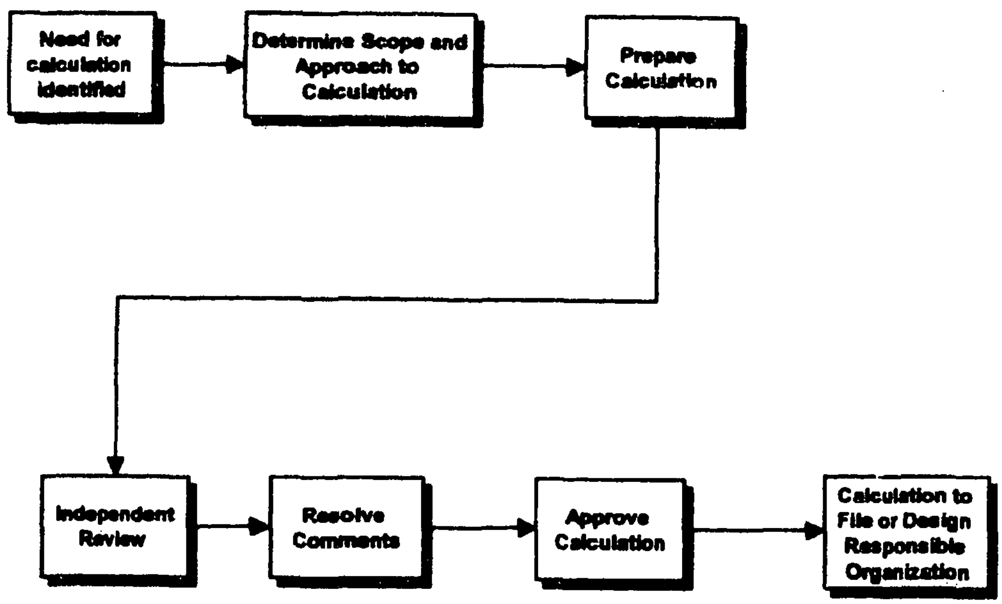
Like RT&C, Controlled Documents (CD) readily identifies to the user when outstanding changes exist against the current revision of a document. When a document has been checked out for use in the field, CD automatically notifies the user when a new revision is issued.

RECENT/PLANNED IMPROVEMENTS

Various legacy databases from AE and ComEd records are being migrated 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 17
Calculation Process
NEP-12-02



Calculation Process

NEP-12-02

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.

Drawings	Codes
Design criteria	Standards
Applicable previous calculations	Studies
System descriptions	Commitments to Regulatory Agencies

Adequately document Engineering Judgment, if applicable, to permit Reviewer to verify logic.

Once the calc is completed, the calc may be checked prior to being submitted for an independent review.

After all comments generated through the independent review have been resolved, the calc 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 based on their training, experience, and level of skill The Reviewer shall have had no influence on inputs or approaches utilized in the design development The Reviewer is responsible to ensure the calculations

Completeness	Meets applicable codes
Technical adequacy	Meets applicable standards
Accuracy	Meets quality requirements
Appropriateness for stated purpose	Meets licensing commitments
Appropriateness of assumptions	Reasonableness of 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 any differences will not affect the comparison and that conclusions are consistent.

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

RECENT/PLANNED IMPROVEMENTS

As stated in the November 12, 1996 letter from T.J. Maiman to A. Bill Beach, ComEd intends to take the following action

Critical calculations are an important part of maintaining the Design Bases. ComEd will define the set of calculations that are critical to maintaining design control and reconstitute them when they do not exist. Until this long term program is completed, we will validate or reconstitute a critical calculation when needed to support ongoing operations or new modifications.

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 separat[illegible]ice 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 fue[illegible]signs 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 Control 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 (10CFR50.59 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 10CFR50.59. 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 10CFR50.59.
- 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)

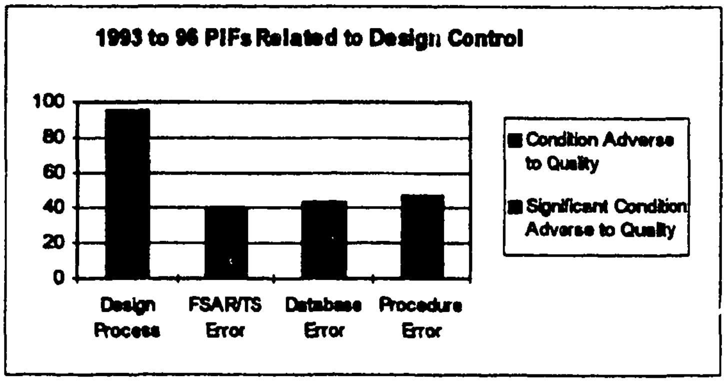
A review was performed of NFS generated PIFs from 1993 (the first year the PIF process was used in NFS) to present (November 8, 1996). As described in Action (d), the PIF process is common to all six nuclear stations and is also used by NFS to identify, document, assess, and correct design bases and 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."

[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".

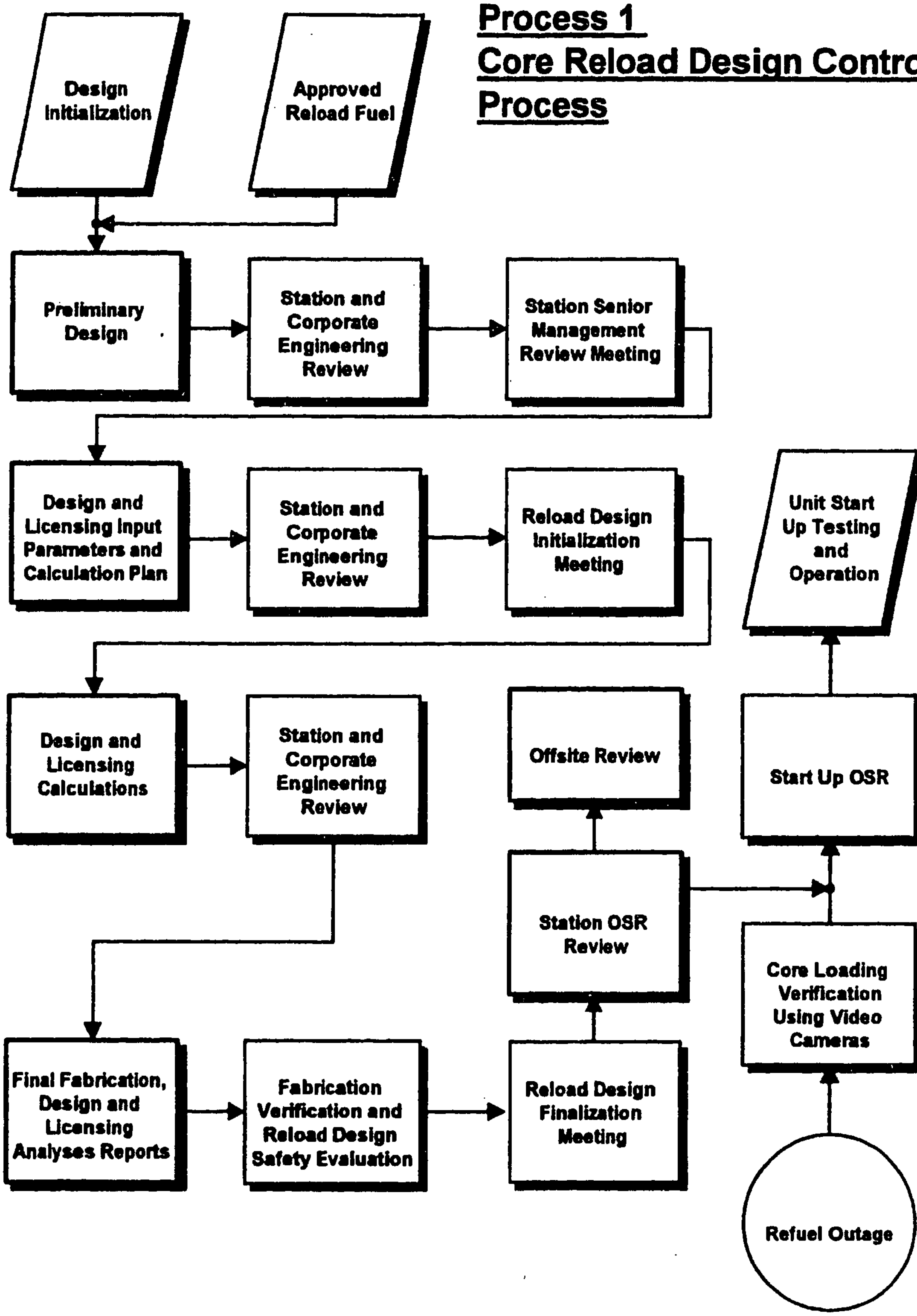
However, weaknesses were also identified such as:

> "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."

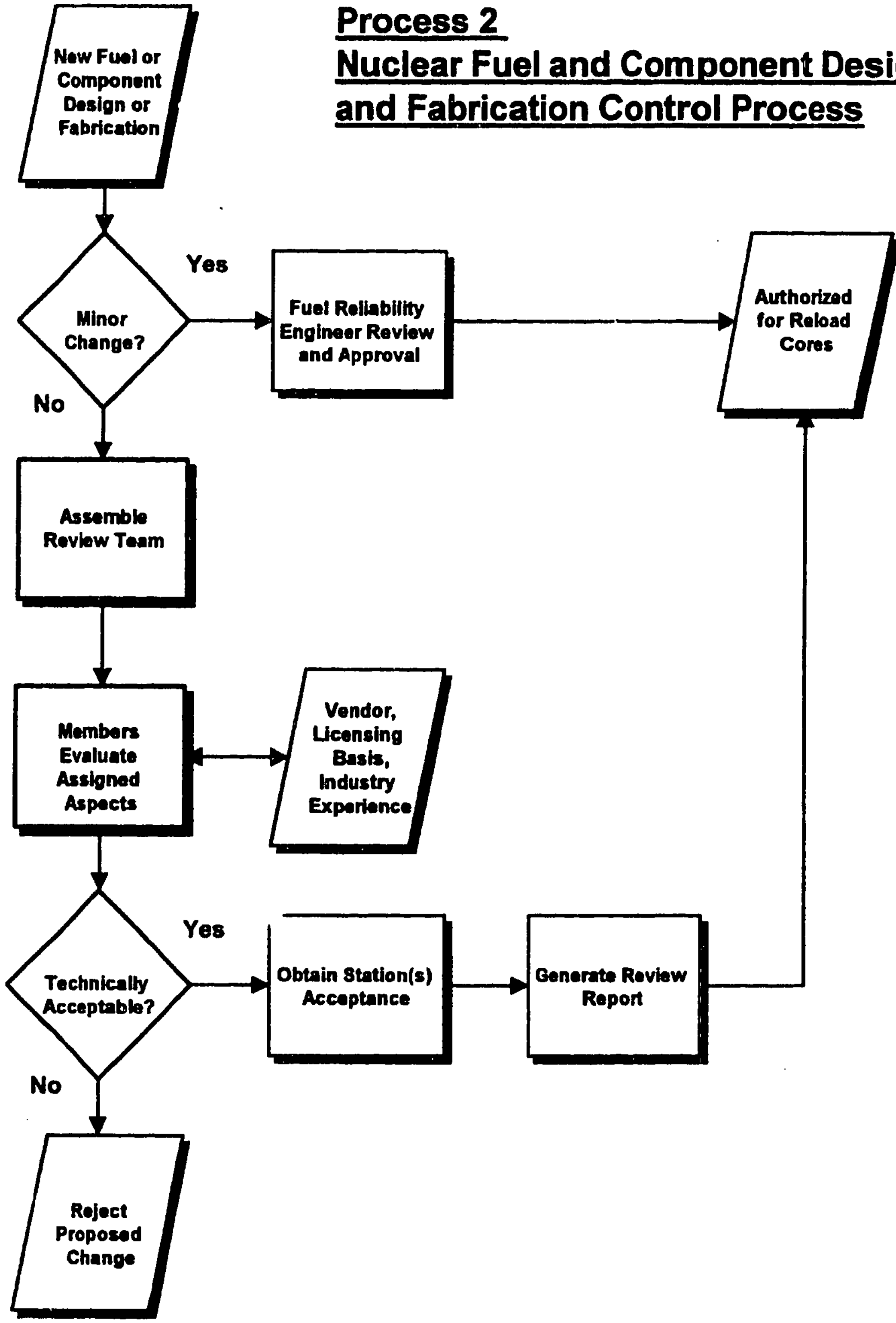
These weaknesses have been and are continuing to be addressed through enhancements to the reload design process.

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