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ComEd

February 6, 1997

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

Subject: Dresden Station Units 2 and 3
Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Basis Information
NRC Docket Numbers: 50-237 and 50-249

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) J. S. Perry letter to A. B. Beach dated November 8, 1996

(c) 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"

(d) J. S. Perry letter to H. L. Thompson dated January 13, 1997, "Interim Response to Dresden Independent Safety Inspection Report"

(e) 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"

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This letter transmits Dresden 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, ComEd concludes for that Dresden Station there is reasonable assurance that its procedures reflect the design bases and that the plant is configured and operated in a manner that is substantially consistent with the design bases, as defined in 10 CFR 50.2, or as otherwise permitted under the NRC's regulations and that discrepancies involving the

design bases are resolved appropriately. Moreover, a corrective action program is in place to resolve any deviations that may be identified from time to time.

Dresden Station's process for developing its response was structured to take a comprehensive look at the configuration management program as it applies to the design bases. A team comprised of over 50 personnel participated in the preparation or review of this response. It included and reported to a site team leader. Team members were chosen in part based on their knowledge of design basis issues and programs that involve site design bases. The team leader reported to a corporate leader who provided in-depth reviews for consistency and completeness across the six, ComEd nuclear stations.

Validation and review of the response was conducted at several levels. The team established a process for validating and documenting the sources of all documents relied on as input into the response. Subject matter experts wrote individual sections of the attached response. Each of the sections was reviewed by least one knowledgeable individual other than the preparer. Senior management reviewed the entire response. Dresden's Onsite Review reviewed the entire response. This review concluded: 1) The individual section reviews were adequate; 2) The overall conclusions were appropriate and adequately supported; and 3) Statements of ongoing and future planned actions were appropriate. An external review team (comprised of experienced individuals who have extensive experience with the nuclear regulatory process and who have only limited involvement in ComEd's day-to-day activities) was assembled to provide an independent assessment of the quality, completeness and responsiveness of the reply. Finally, Dresden's Plant Operations Review Committee (PORC) reviewed the document. These processes for preparing and reviewing the response, especially the internal checks and balances that were built in at each level of the process, provide a high level of assurance of the completeness and accuracy of the response.

The station's response is structured around the five action items in the 50.54(f) request, as well as the request regarding "Design Review and Reconstitution Programs." The attachment to this letter is supplemented with three appendices:

- Appendix I, "ComEd Organization Restructuring to Improve Dresden Station's Ownership and Control of the Design Bases," discusses the three-year plan to establish a ComEd design engineering organization. This appendix also discusses other supporting roles of 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, processes deployed at Dresden Station. This appendix supports Action (a) directly.

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

Current Situation

Dresden Station's conclusion that there is reasonable assurance that the plant is substantially configured and operated consistent with its design bases is based on several factors. First, systems were thoroughly tested to demonstrate design basis conformance during pre-operational testing prior to initial startup. Since then, changes to the plant's physical configuration and operating procedures have been made in accordance with programs that were designed and adopted to assure continuing consistency with the design bases. Under those programs, changes to the plant's configuration and its operating procedures are subject to appropriate Onsite and Offsite Reviews. ComEd has an accredited training program for all critical processes. These programs have been improved and upgraded over time.

Also, over the operating life of the plant, Dresden Station also has generally responded as expected during transients. Where operational issues have been discovered, Dresden has taken actions to resolve them. Self-assessments, ComEd audits, NRC inspections and third party reviews have repeatedly probed the implementation of programs designed to maintain the design bases, the status of plant equipment, and the adequacy of plant procedures. Where substantial discrepancies have been identified: 1) their root causes and extents of occurrence have been determined; 2) the discrepancies have been corrected; and 3) the processes that permitted them to occur have been strengthened by eliminating their underlying root causes.

ComEd acknowledges that programmatic weaknesses still exist in Dresden's design control processes. The recently completed NRC Independent Safety Inspection (ISI) identified significant programmatic weaknesses, especially in the area of calculation control. The NRC ISI performed a detailed examination of three significant Dresden safety system, and examined two other systems in lesser detail. Design basis issues were identified in all five systems. In response to the issues raised by the ISI, Dresden committed to certain corrective actions (References (b) and (d)).

ComEd has since performed a series of reviews to reverify, in light of these findings, that current plant conditions are safe and support continued operation. This included a prompt evaluation of current system surveillance and acceptance criteria for twelve risk-significant systems. In each case, it was determined that surveillance results demonstrate that the equipment operates as expected and that the system will perform its safety function.

These prompt reviews, in conjunction with all of the considerations discussed in the attachments, support the conclusion that there is reasonable assurance that the plant is

currently configured and operated in a manner that is substantially consistent with the design bases. In addition, other recent NRC inspections (Engineering and Technical Support, and the Motor Operated Valve program) resulted in no items of non-compliance. In the interim response to the ISI (Reference (d)), we noted that we are in the process of developing a comprehensive set of actions to address the deficiencies identified by the ISI, and intend to provide the NRC with a final response describing those actions by the end of February.

Future Action

ComEd recognizes that despite concluding that there is reasonable assurance that Dresden is configured and operated in a manner that is substantially consistent with its design bases, recent inspections have highlighted weaknesses that need to be addressed at all ComEd Stations. Following recent inspections at Zion Engineering and Technical Support (E&TS), LaSalle Service Water Operational Performance Inspection (SWOPI), and Dresden (ISI), Mr. Thomas J. Maiman (Chief Nuclear Officer) communicated to Mr. A. Bill Beach (NRC - Region III) on November 12, 1996, (Reference (c)), that all six ComEd Nuclear Station would implement thirteen actions that would enhance ComEd's assurance that the nuclear stations are operated and maintained consistent with their design bases. These actions, in addition to the review of twelve risk-significant systems described above, included: 1) conduct an Engineering Department Safety System Functional Inspection; 2) establish an Engineering Assurance Group; 3) commence consistency reviews of Inservice Testing programs against the design bases; and, 4) conduct Quality Assurance audits of major contractors in the areas of design control. In addition to these actions, ComEd also committed to improve the quality, retrievability and consistency of Design Basis Documents (Reference (e)).

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. Some of those commitments have been discussed in this response for continuity. Also, this response contains numerous descriptions of current processes. The processes, as described in this response, will change over time as the required improvements are made. To clarify, ComEd's commitment to future actions regarding the quality, maintenance and accessibility of design bases, we have provided those commitments under separate cover to the NRC.

In conclusion, ComEd is dedicated to the safe operation of its nuclear power plants. We clearly recognize the importance of operating and maintaining station in conformance with design bases. The commitments in the referenced letters, and the ongoing oversight roles by Site Engineering Assurance and Quality Verification groups, and Corporate Engineering Assurance, Chief Engineering, and Quality

Verification Groups all contribute to enhance the current reasonable assurance that the stations are 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

J. Stephen Perry
Site Vice President
Dresden Station

TJM/tb

Attachment

cc: A. B. Beach, Regional Administrator - RIII
J. Callan, Executive Director for Operations
S. Collins, Director - NRR
J. Stang, Dresden Project Manager - NRR
C. Vanderniet, Senior Resident Inspector - Dresden
Office of Nuclear Facility Safety - IDNS

COUNTY OF DuPage
STATE OF Illinois

AFFIDAVIT

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

Thomas J. Maiman

Thomas J. Maiman
Executive Vice President
Chief Nuclear Officer

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

My commission expires 5-19-98.

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

Michelle Kowalkowski

Notary Public

DRESDEN STATION

RESPONSE TO NRC REQUEST FOR INFORMATION
FEBRUARY 6, 1997

Adequacy and Availability of Design Bases

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EXECUTIVE SUMMARY

The following provides an overview of the Dresden Station 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): There are Dresden Station and related corporate engineering design and configuration control processes and programs currently in place, including those that implement 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR Part 50. The processes and programs are described in detail in this response in Action (a), as well as in Appendix II. As detailed in the response, Dresden's processes and programs implement regulatory requirements for maintaining the configuration of the plant and the design documentation. The results of our review support a finding that the scope and extent of these processes are adequate to maintain the plant configuration and operation consistent with the design bases. We have identified improvements in our processes and programs, and we intend to implement these improvements to further strengthen our controls over the configuration control program.

Action (b): An adequate basis exists for concluding that design bases requirements are translated into operating, maintenance, and testing procedures. Procedures have been continually reviewed and improved over the operating life of the station, reflecting both the station and industry experience. Based on the formal checks and balances provided in the procedure update and revision process, qualification of the procedure reviewers, and the results of past audits, inspections and assessments, including the resolution of identified problems, we conclude that there is reasonable assurance that design bases requirements are adequately reflected in our operating, maintenance and testing procedures. We have identified discrepancies during plant operation, and these discrepancies have been resolved in accordance with our corrective action program.

Action (c): An adequate basis exists for concluding that there is reasonable assurance that system, structure, and component configuration and performance at Dresden Station are substantially consistent with the design bases. The rationale is supported by:

(1) original preoperational testing;
(2) procedural control of configuration changes, maintenance actions and operational changes to maintain consistency with the design bases;
(3) configuration and performance verifications through plant walkdowns, testing and plant operational control;
(4) special verifications and programs that have improved access to design basis information and enhanced its control and use;
(5) special programs that have confirmed the equipment configuration and/or capability of equipment to function during seismic or accident conditions; and
(6) self assessments, audits and inspections of design and configuration control (these assessments, audits and inspections have identified documentation and plant deficiencies that have been resolved as part of our corrective action program).

Dresden Station specifically recognizes that the NRC Independent Safety Inspection team in 1996 concluded that certain weaknesses exist in the Dresden configuration management process. Moreover, timeliness of completing programs and resolving issues has been a recurring weakness. Dresden Station has taken action to reverify that current plant conditions are safe and support continued operation, as well as action to verify that backlog issues will not raise operability or design bases conformance issues.

We also have future actions planned to continue to address improvements in our configuration management program. These improvements will facilitate access to design bases information and improve management of the configuration consistent with the design bases.

Action (d): There are processes in place at Dresden Station for identification of problems and implementation of corrective actions, including actions to determine the extent of problems and to prevent recurrence. There are also processes in place addressing reporting to NRC. In total, Dresden's problem identification and corrective action processes are capable of identifying, correcting and preventing the recurrence of any significant nonconformances with the plant design bases. Specific weaknesses that have been identified in our processes have been corrected.

Action (e): We have conducted an assessment of the overall effectiveness of current processes and programs in concluding that the configuration of Dresden Station is consistent with the design bases. Although deficiencies associated with the implementation of the design and configuration control processes and programs have been identified at times, we have implemented effective corrective actions that have resulted in upgraded configuration controls. Recent audits and assessments continue to reinforce the need to upgrade our station processes and improve station operations. ComEd is committed to perform additional activities which will further enhance confidence in Dresden's conformance to its design and licensing bases.

Overall, based on our review of the processes and programs in place to ensure consistency with the plant design bases, and implementation of our corrective action program, we believe that Dresden Station's processes and programs are effective overall in maintaining the plant consistent with its design bases. Future actions to be taken regarding the quality and accessibility of design bases information, design control, configuration management and related areas are summarized in Section 5.0 of the response. These activities are described in the January 30, 1997 T. J. Maiman letter to A. B. Beach.

Dresden Response to 10 CFR 50.54(f)

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Dresden 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, as discussed in Appendix I, which is followed at both the corporate office and the sites, including Dresden Station.

The complementary configuration management roles of the corporate office and site Engineering Departments are discussed in Appendix I. In the ComEd corporate office, implementation of the configuration management model is the responsibility of the Chief Engineer, Configuration Management, who reports directly to the Engineering Vice President.

The corporate office is responsible for Nuclear Engineering Procedures (NEPs) and Nuclear Station Work Procedures (NSWPs). NEPs and NSWPs provide guidance on corporate expectations for configuration control processes. Dresden is responsible for the administrative procedures that implement the processes which constitute the configuration management model. These administrative procedures specify how work is to be performed and how the station is to be operated to assure consistency with the design bases.

The major elements of the engineering design and configuration control processes are summarized in Appendix II, and include both corporate and station procedures. The matrix also summarizes the processes for implementing 10 CFR 50.59 and 10 CFR 50.71(e), and illustrates how the processes relate to the configuration management model.

The procedures for engineering design and configuration control are structured to achieve the following objectives:

- Assure the establishment of 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 design changes continue to satisfy design basis requirements, through controlled processes for review and approval of the design changes, installation, testing and operation.
- Assure compliance with 10 CFR 50.59.
- Assure implementation of the FSAR update requirements in 10 CFR 50.71(e).
- Assure that Quality Control (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.

- Assure that procedures reflect the current design and correct plant configuration.
- Assures that procedure preparers and reviewers to have ready access to design basis information and assure that they are familiar with the design bases.
- Assure that personnel are trained.

As with all procedures, procedural adherence is a clearly communicated management expectation and processes are in place to reinforce adherence. Nuclear station and corporate personnel are required by ComEd policy to comply with and adhere to work procedures, or, if they cannot comply, they must stop work, secure the station or equipment in a safe condition, and seek clarification from their supervisor.

Appendix III discusses the Nuclear Fuels design process.

1.2 Requirements of Procedures Which Control Design Bases

Dresden's procedures for implementing the configuration management model address four principal areas: 1) design control -- determines the impact of the proposed actions on and assures consistency with the design bases; 2) licensing basis review -- determines that the impact of the proposed actions are consistent with the licensing basis documents, e.g., UFSAR; 3) safety evaluation under 10 CFR 50.59 -- determines whether a proposed change involves an Unreviewed Safety Question (USQ); and 4) configuration control -- assures that documentation is updated in a timely manner after a change is made. The following overviews in each of these four areas summarize the important steps taken to maintain the configuration and operation of Dresden Station consistent with its design bases.

The design control processes conform to Criterion III of Appendix B and include the following provisions:

- The procedures apply to new design work and design changes for structures, systems, and components;
- Design work is reviewed for conformance with design bases (or appropriate changes are implemented in the licensing basis);
- Design work is documented in calculations, analyses, specifications, drawings, or other controlled documents;
- Design work is subject to design verification;
- Design work is approved by management; and
- Design changes are reflected in controlled sets of analyses, specifications, and drawings.

The procedures for implementing Section 50.71(e) include the following provisions:

- Station 10 CFR 50.59 safety evaluations are reviewed to determine whether the FSAR needs to be updated;
- Effects of safety analyses for license amendments are incorporated in FSAR updates;

- Effects of other safety analyses required to be submitted to the NRC are incorporated in FSAR updates;
- Updates to the FSAR include not only changed information but also new information and analyses as identified above;
- Between updates, identified changes for the FSAR are controlled and accessible to plant personnel; and
- Responsible personnel receive training in the above procedures.

The procedures for implementing 10 CFR 50.59 include the following provisions:

- Evaluations are performed for the following: changes to the FSAR and Technical Specifications, changes to design and operation, procedure changes, temporary modifications, and prolonged operation with degraded and nonconforming conditions;
- Changes are screened to determine whether they involve a change in the FSAR or Technical Specifications;
- Safety evaluations pursuant to 10 CFR 50.59 are documented;
- Safety evaluations are subject to review and approval, and, on a selected basis, review by the Plant Operations Review Committee (PORC);
- Unreviewed Safety Questions and changes to the Technical Specifications are submitted to the NRC for approval as part of a license amendment application; and
- Responsible personnel receive training on the above procedures.

The configuration control procedures include the following provisions:

- Prior to approval, design changes are evaluated for conformance with design bases;
- Changes are verified to be consistent with the design bases;
- Prior to approval, design changes are evaluated to determine their impact upon operating, maintenance, and testing procedures and training programs, and appropriate changes are made to affected procedures and programs;
- Approved design changes are implemented in accordance with controlled documents, e.g., work packages, installation procedures, or specifications;
- Modifications are subjected to QC inspections and post modification tests;
- Temporary Alterations are evaluated and are subject to engineering approval;
- Changes to operating, maintenance, and testing procedures are reviewed to determine their conformance with the design bases and other design documents;
- Changes in the plant systems as described in the UFSAR are reviewed under 10 CFR 50.59 to determine whether an Unreviewed Safety Question exists; and
- Responsible personnel receive training in the above procedure(s).

1.3 Overview of Processes Which Implement Design and Configuration Control

Processes which implement Design Control:

- Work Initiation

- Work Planning and Design
- Temporary Alteration

Processes which implement Configuration Control:

- Work Execution
- Design Document Update

Each of these processes are 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. Repetitive scheduled maintenance activities such as oil changes on equipment may be initiated by a predefined Electronic Work Control System (EWCS) generated work request. Other issues such as door handles, bulb replacements, etc., do not require an AR. For engineering assistance and evaluation, the Engineering Request (ER) is used. For problem investigation and corrective action, the Performance Improvement Form (PIF) (discussed in Action (d)) is used.

ARs and PIFs are reviewed by the shift manager or a licensed RO for impacts on operability and Technical Specifications. If immediate action is required, actions are promptly implemented. If a piece of equipment important to safety has been identified as being degraded, an operability assessment is performed. An experienced, multi-disciplinary screening committee assigns 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 design change (Section 1.5), a procedure change or new procedure (as described in Action (b)), use of other than "like-for-like" replacement parts (Section 1.9), or a setpoint change (Section 1.10). An Engineering Request is generated to request engineering to prepare the design change.

Work analyst responsibilities may include a verification walkdown prior to starting the work package to confirm that the configuration of the plant is consistent with the design documents. Any discrepancies between the plant configuration and the design documents are brought to the attention of the Engineering department. In this case, an ER or a PIF is used to document and resolve plant discrepancies.

Replacement parts are evaluated by the "Parts and Materials Management" personnel under a "like-for-like" replacement process in accordance with station procedures. If the work analyst identifies that a "like-for-like" part is not available, an ER is issued for a non "like-for-like" replacement evaluation (see Appendix II, Process 8).

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 Temporary Alteration (Temporary Modification)

Temporary Alterations (Section 1.6) are used to document the acceptability of an interim change to the plant configuration. A safety evaluation (Section 1.7) may be required depending on safety significance and the time required to complete the permanent corrective action.

Consistency of operation with the design bases pending completion of work is assured by performing evaluations and/or taking compensatory actions and/or documenting the temporary condition as a Temporary Alteration.

1.3.4 Work Execution

Actual work, routine maintenance, and design changes are executed with a work package prepared by the Work Control Department. Work packages which modify the plant are based upon a design change which is approved by engineering. Work is controlled by an existing procedure or by a work package prepared under the work control process. If design changes cannot be installed as designed, changes are documented on a field change request and reviewed by engineering. Post maintenance testing helps ensure that work was completed properly and that equipment conforms to applicable design requirements and can be returned to service. If a special test is required, it is prepared using the special test procedure.

The Out-of-Service (Section 1.11) process ensures that operational plant configuration is controlled consistent with the design bases during performance of maintenance activities. Technical Specification surveillances, which verify design requirements, are also controlled through the work control process using controlled procedures. Any discrepancies are reported on a Performance Improvement Form (PIF); PIFs are distributed and incorporated into the PIF database.

Consistency between operation and the plant's design bases is maintained by the Work Control Process through the Return-to-Service process, update of EWCS data, completion of design document changes, etc., during the closeout process.

1.3.5 Design Document Updates

Design document changes may be required either because of a deficiency, such as a deviation between a document and the as-built condition of the plant, or as the result of a design change. The Document Change Request (DCR) (Section 1.13) process is used to control design document changes. Other update processes apply to the UFSAR (Section 1.8), and station procedures (Action (b)). Vendor manual and/or operating or maintenance procedures may need to be

changed based on information received from a vendor as part of the Vendor Equipment Technical Information Program (Section 1.12).

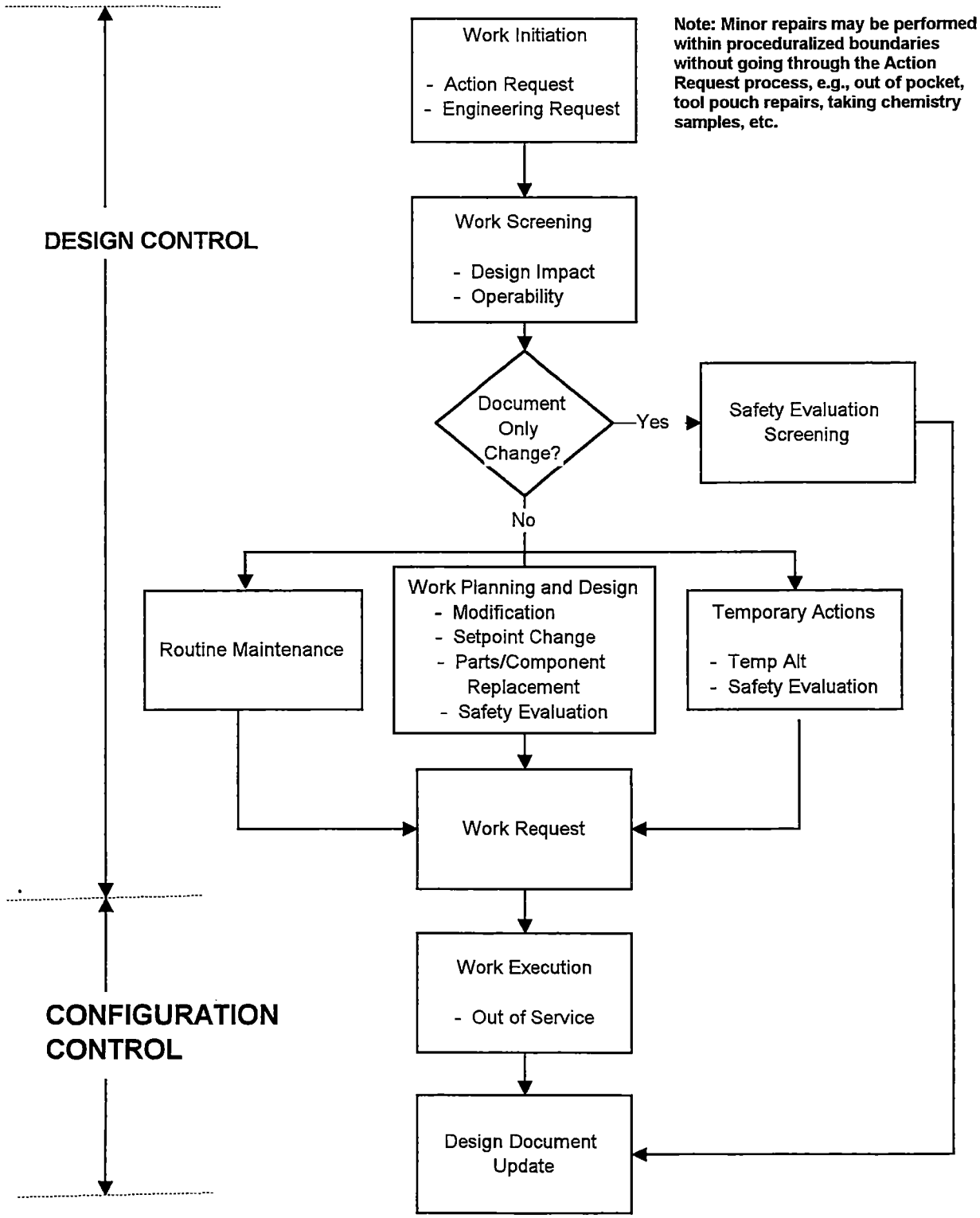
New or revised calculations (Section 1.16) may be required at various stages of different processes. Some calculations are prepared using computer software that is controlled by the Engineering Software and Revision Process (Appendix II, Process 11).

Design Basis Documents (DBDs) (Section 1.15) which describe the system design have been prepared for twenty of the plant's systems and three topical areas and are maintained current by the design control process.

Configuration control, accessibility, and retrievability of design basis information and change documents have been enhanced through the use of the Electronic Work Control System (EWCS). All design changes are processed and tracked through EWCS which provides ready access to information such as Engineering Change Notices (ECNs), Field Change Requests (FCRs), equipment data and drawings which were associated with a particular design.

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

Typical Structure of Design and Configuration Control



1.4 Work Control Process

1.4.1 Action Request Initiation

The work control and work request processes at Dresden are designed to allow the plant to be operated and maintained while controlling and maintaining the design basis. A combination of station and corporate procedures are in place to control the work process.

The current process for requesting performance of a work activity is to initiate an Action Request (AR). The AR process is intended to provide all site personnel with an easy and accessible process to identify a physical deficiency or discrepancy associated with any structure, system, or component. In some instances, a Performance Improvement Form (PIF) may also be initiated and processed as described in Action (d). Action Requests are subjected to a screening process which includes an assessment of the urgency, type of work required, and prioritization.

Appendix II, Process 1 describes the corporate process for screening ARs. If, in the opinion of the screening committee, any aspect of the requested work could potentially impact the design bases of the plant or the Technical Specifications, an Engineering Request (ER) is initiated to require engineering involvement. The screening committee assessment does not constitute a final judgment. If, in the process of developing a work package, the analyst or any of the reviewers determine that an activity potentially affects the design basis, the development of the package is stopped and engineering assistance is requested. Engineering reviews all ERs from the work analysts and determines whether work can continue through a Work Request or whether a design change or other engineering output document is required to support the work package.

1.4.2 Work Package Preparation

Once the AR is prioritized and approved, a Work Request package is generated. This process is governed by a Dresden Administrative Procedure. Depending upon the nature of the work, a work package may receive reviews from several departments. Work packages are prepared and reviewed by experienced Work Planning personnel.

During preparation, a work package preparer reviews the requested work for impact on various programs which comprise the design bases of the plant, e.g., Fire Protection, Inservice Inspection/Inservice Testing, Foreign Material Exclusion, Vendor Equipment Technical Information Program, and impact on Safety-Related, Regulatory-Related, Environmentally Qualified equipment, Security Systems, and Control Room equipment. The Work Package Preparer ensures that a 10 CFR 50.59 Screening/Safety Evaluation has been completed if required. Work Packages for normal repair and maintenance activities generally do not require a 10 CFR 50.59 Screening/Safety Evaluation. Design Changes, Temporary Alterations, Setpoint Changes, and the like are evaluated per 10 CFR 50.59 under the applicable process. Work packages resulting from these processes do not require additional 10 CFR 50.59 review. The Safety Evaluation process is described in Appendix II, Process 7 and discussed in Section 1.7 below.

A multi-disciplined technical review of the Work Package is generally not required, except in specific cases delineated in the Dresden Administrative Procedure. Technical Reviews are performed in accordance with detailed guidelines to ensure the technical adequacy of the Work Package. Dresden's Technical Specifications require that Technical Reviewers meet the applicable experience requirements of Sections 4.2 and 4.4 of ANSI N18.1-1971. The review of Work Packages associated with the licensing/design bases includes following questions:

- What is the impact on the system response or margin?
- Does a procedure change affect any Technical Specifications/UFSAR?
- Does the change render any other equipment inoperable which has NOT been identified?
- Does the change adequately restore the system to an operable status?
- Have all required design inputs been considered?
- Have all applicable drawings have been reviewed for potential system interaction?
- Are all calculations correct?
- Is the 10 CFR 50.59 Screening/Safety Evaluation, if applicable, accurate and complete?

1.4.3 Work Package Approval

A Work Package Approver performs the following:

- Verifies that the work instructions are adequate and technically correct for the scope of work;
- Verifies that the Work Package complies with station procedures, Technical Specifications, applicable codes, the UFSAR, etc;
- Verifies that technical review and any cross-discipline review(s) are complete (if required), or concurs that a formal technical review is not required; and
- Verifies that a 10 CFR 50.59 Screening or Safety Evaluation was completed or that one was not required.

Once a package has been developed and approved, the work is scheduled according to plant need and plant status. When the package is ready to be worked, it is coordinated through the Work Week Manager. The Work Week Manager functions to coordinate work activities with Operations, Radiation Protection, and Maintenance and provides an additional check on the type of work being done and the impact on plant operation.

1.5 Design Change Process

A corporate overview of the Design Control Process is described in Appendix II, Process 2. A Roadmap to the Design Change Processes is described in Appendix II, Process 3. The Design Change process at Dresden Station is controlled through a modification program which uses two processes which are structured after the corporate procedure described in Appendix II, Process 4, and that supplement the corporate procedure. These two Dresden processes are the Modification and Exempt Change processes. The determination of which process is used is based upon the

content and complexity of the design change. (Previously, a Minor Modification process was used. A few older design changes processed as Minor Modifications are still open. These modifications will be completed as Exempt Changes.) There are several specific areas in these processes where there is review against the design bases. The design change process requires that the station Simulator be maintained current with the plant design. Additionally, detailed checklists are utilized throughout the process to ensure that changes to configuration management documents such as drawings, procedures, UFSAR, and Design Basis Documents are addressed and that required training, e.g, Maintenance and Operations is provided.

An Engineering Change Notice (ECN) is used to communicate design changes included in the Design Change package. ECNs are required for Modifications and are utilized as necessary for Exempt Changes. The ECN process is described in Appendix II, Process 12.

Dresden station's procedures have been specifically enhanced as a result of past weaknesses encountered. For example, a lack of early involvement from other departments, e.g., Operations, Maintenance, Training, was identified. In response, several changes were made to ensure that affected departments are involved throughout the design process. These enhancements include the following:

- Expectations and checklists were developed to ensure that departments participating in the Design Change process were aware of their departments involvement and their required input to the process;
- Additional mandatory design review meetings (conducted when designa are approximately 30%, 50%, and 70% complete) were instituted; and
- Scope meetings are now conducted for all Exempt Changes.

Another weakness in the Design Change Process was that too many Field Change Requests (FCRs) were being initiated during installations. In response, an FCR threshold was established which when reached, forces the design to be reviewed for adequacy. A final ECN review meeting is conducted in an effort to reduce the number of FCRs. Dresden's previous backlog of 238 old open design changes was also a concern. Having open changes makes it more difficult for personnel to determine the latest design basis information. In June 1995, efforts were initiated to reduce the number of old open design changes. As of January 1997, the number has been substantially reduced to 21.

An additional deficiency in this process was that wiring diagrams did not match what was installed in the plant. The Design Change Process now requires wiring verifications to be performed through walkdowns of affected electrical cable terminations, panels, and cabinet wiring. A Pre-Wiring Verification is performed to ensure that as-designed drawings reflect the actual installed conditions for all terminal points which will be utilized for the design change. A Post-Wiring Verification is also performed, to assure that the wire-by-wire revisions have been adequately translated from the design drawings to the physical components, and to verify that no changes from other design changes have occurred that would affect the testing.

1.6 Temporary Alteration Process

Temporary Alterations (TAs) are interim changes to the approved design configuration of a structure, system, or component. These alterations do not circumvent the process for permanent plant design changes. The TA Process at Dresden is controlled by a Dresden Administrative Procedure which is described in Appendix II, Process 6.

In 1995, the Station's Event Screening Committee raised a concern about an adverse trend in the implementation of the station's Temporary Alteration Program. At that time, over 140 TAs were installed in the plant. This situation resulted in Design Engineering taking ownership of the program and assigning a dedicated Design Engineer as the program owner. A comprehensive effort was launched to review all identified plant alterations and create awareness among the site personnel about the Temporary Alteration Program. As a result of this effort, the number of installed Temporary Alterations was reduced from 140 to 23, as of December 1996.

Heightened station awareness of the Temporary Alteration program also resulted in three times more PIFs being written in the second half of 1995 concerning possible undocumented plant alterations as compared with the first half of 1995. This was a good indication of plant personnel identifying historical deficiencies and correcting them. Assigning ownership resulted in proper screening of the issues and enhanced control in facilitating the resolution of issues through regular plant maintenance/modifications rather than a Temporary Alteration.

1.7 Safety Evaluation Process

Currently, 10 CFR 50.59 Screenings and Safety Evaluations are prepared utilizing a Dresden site procedure which conforms to the guidelines and requirements of ComEd's corporate procedure NOD TS-11. (The site procedure will be replaced by a new corporate procedure, NSWP-A-04.) The current 10 CFR 50.59 process and planned changes are described in Appendix II, Process 13.

10 CFR 50.59 Screenings and Safety Evaluations provide the basis for determining whether a Procedure Change, Test, Experiment, or Facility Change could:

- Make changes to the facility as described in the UFSAR;
- Make changes to procedures as described in the UFSAR;
- Conduct tests or experiments which are <u>NOT</u> described in the UFSAR; and
- Involve a Unreviewed Safety Question or a change to the Technical Specifications.

1.8 UFSAR Update Process

Changes made to the facility, equipment, analyses, procedures, programs, or organizations which change the description included in the UFSAR, require that a UFSAR change be initiated. UFSAR changes are controlled through detailed preparation and review processes. This process is defined by a station procedure and is discussed in Appendix II, Process 19.

Changes to the UFSAR can result from design changes. Alternatively, they can be self-generated as part of a general UFSAR update program or a UFSAR review conducted through the normal self-assessment process. A 10 CFR 50.59 Safety Evaluation is performed for all non editorial UFSAR changes. Editorial changes may be screened. The safety evaluation provides an important checkpoint in the process to ensure regulatory compliance and maintain design control. All UFSAR changes are reviewed and approved by a cognizant Engineering Supervisor. The Onsite Review also provides an additional final checkpoint in the process.

1.9 Parts and Material Replacement Process

Procurement of parts and materials for maintenance activities is an integral part of work package preparation. The need for parts is identified in the form of a Bill-of-Materials within the Electronic Work Control System (EWCS) which is conveyed to the material control group. Inventories at all ComEd facilities are reviewed to determine availability; if the part is not in stores, it is procured from a vendor. If an exact replacement is not found or cannot be procured, then an evaluation is performed to confirm acceptability with the design of the component or system. This evaluation is performed by experienced individuals using specific procedures and checklists. This process is described more fully in Appendix II, Process 8.

Effectiveness of this program has been the subject of several assessments. See Action (c), Section 3.4.5 for further information.

1.10 Setpoint Change Control Process

Currently, setpoint changes are performed utilizing a Dresden site procedure which conforms to the guidelines and requirements of ComEd's corporate guidance, NOD-MA-10. The process is described in Appendix II, Process 9.

The setpoint change program is designed to assure that all relevant design considerations are addressed prior to making changes. The impact of setpoint changes upon Dresden Station's design bases and licensing documents is controlled through detailed preparation and review processes.

Setpoint changes are prepared by experienced personnel. During preparation of the change, the preparer reviews the setpoint change for impact on Technical Specifications, the UFSAR, Dresden's Design Basis Documents, the Fire Protection Program, procedures, drawings, calculations, and transient and LOCA analysis input parameters.

Setpoint changes receive a 10 CFR 50.59 Screening/Safety Evaluation to provide the basis for determining whether the setpoint change could involve an Unreviewed Safety Question or a change to the Technical Specifications.

In mid-1996, a weakness in the setpoint change control program at Dresden was identified. The process did not allow a setpoint change to be routed to all affected personnel. The procedure was

revised to have setpoint changes controlled and routed in the same manner as Design Changes, through the Electronic Work Control System.

1.11 Out-of-Service Program

This process defines the control procedures used to remove equipment from service. In general, station personnel may initiate an Out-of-Service (OOS) request to enable them to safely perform work on station equipment or to otherwise maintain and control abnormal configurations. This OOS process leads to hanging equipment cards to control operation of the equipment. The process is managed through the Electronic Work Control System. Details concerning the OOS process are provided in Appendix II, Process 20.

A strength of this process is in the many provisions for verifications. Roles and responsibilities of the plant personnel are delineated in the controlling procedure. Training for each work group that interfaces with the OOS process is required. Follow-up training is provided as deemed necessary by the Department Training Coordinator.

Dresden is planning to participate in the six station ComEd standardization of the OOS process. This new process will include a new Electronic OOS program.

1.12 Vendor Equipment Technical Information Program (VETIP)

VETIP at Dresden is designed to control vendor technical information used for the installation, maintenance, operation, testing, calibration, troubleshooting, and storage of equipment. All vendor manual information is processed through the Station's VETIP Coordinator. The process is controlled through a corporate procedure which is supplemented by a station procedure. It is described in Appendix II, Process 14.

Effectiveness of this program has been the subject of several assessments. See Action (c), Section 3.6.1.4 for further information.

1.13 Document Change Request (DCR) Program

The Document Change Request (DCR) process is used to control the incorporation of design changes or as-built information into design documents. The DCR program at Dresden is controlled by a station procedure described in Appendix II, Process 7.

Effectiveness of this program has been the subject of several assessments. See Action (c), Section 3.6.1.6 for further information.

1.14 Configuration Control Using EWCS

The Electronic Work Control System (EWCS) is an on-line workflow and database tool used at Dresden. The system is used to monitor status of Engineering Requests, Design Change

Packages, Engineering Change Notices, Field Change Requests, drawings, calculations, and Document Change Requests.

This configuration control function is governed by a corporate procedure and is described in Appendix II, Process 15.

1.15 Design Basis Documents (DBD Process)

The Design Basis Documents (DBDs) describe the design of the plant and are updated as necessary to incorporate changes resulting from various types of modifications. The process for developing and updating DBDs at Dresden is controlled by a corporate procedure, and is described in Appendix II, Processes 10 and 16. The corporate process is also supplemented by a Dresden administrative procedure. The station procedure describes administrative processing of DBD changes and requires that a 10 CFR 50.59 Screening/Safety Evaluation be performed for DBD changes. Additionally, guidance is provided to ensure that other programs and procedures which may be impacted by the DBD change are addressed.

The DBD program addresses the previous weakness at Dresden Station that design bases information was not readily available for use at the site. The DBDs that have been completed for twenty plant systems and three topical areas (which address Seismic, Design Basis Events, and Single Failure Criteria) bring information together and present it in a coherent manner to facilitate effective use.

1.16 Calculations

Calculations are prepared, reviewed, approved, and controlled in accordance with the corporate procedure described in Appendix II, Process 17. Calculation control was reviewed during the 1996 NRC Independent Safety Investigation. This is described further in Section 5.7.1. Calculations are performed using software which is controlled in accordance with the NEP-20 series. These controls are further describe in Appendix II, Process 10.

1.17 Operability Determination Process

Operability Determinations are performed when the capability of a structure, system, or component (SSC) to perform its specified function(s) as required by the Technical Specifications or UFSAR cannot be unequivocally demonstrated, or where a degraded or nonconforming condition results in a judgment that the equipment is operable but that there are remaining concerns or uncertainties. The process for performing Operability Determinations is governed by a Dresden Station procedure which is consistent with NRC guidance provided in Generic Letter 91-18. This process is described in Appendix II, Process 18.

Since the new operability program was initiated in September 1996, eighty-five (85) operability issues have been identified through the Performance Improvement Form (PIF) program.

Of the identified issues:

- Seven (7) are currently in the screening process;
- Forty-two (42) have been evaluated as "No Concern;"
- Nineteen (19) have been determined as "Operable But Degraded" and are in the correction process;
- Thirteen (13) initially confirmed as "Concerns" have been closed; and
- Four (4) were determined as "Inoperable" but were corrected and declared operable, (These included two issues related to the Technical Support Center and Control Room HVAC and two issues related to the Fire Protection system.)

Open "Operable but Degraded" conditions are periodically reviewed by the Plant Operations Review Committee (PORC). If, in the judgment of PORC, there are an excessive number of "Operable but Degraded" conditions, remedial actions are taken. PORC's primary concern is safe unit operation.

Historically, Operability Evaluations were performed primarily by offsite corporate Engineering personnel. When Nuclear Engineering was established onsite as Site Engineering, the same evaluation process was used; however, no satisfactory tracking program or comprehensive operability process procedure existed. The operability process was not well integrated with the PIF process, which relied on engineers and operators to recognize potential operability concerns. Many site personnel did not have the depth of training which had been provided to corporate engineers.

Several actions have been taken to correct these historic weaknesses. A database and tracking system was set up in January 1995 and regular reviews of open Operability Evaluations commenced. Corporate issued a common procedural guidance document regarding performance of Operability Determinations. The new site procedure (described in Appendix II) was developed to incorporate the corporate guidance, reflect administrative program controls, and delineate Operations and Engineering functions. Previously, Operations identification of Operability issues were done via a PIF and verbal notification, while the Operability procedure was restricted to the activities of Engineering. The new procedure was designed to integrate Operations and Engineering activities with a link to the PIF process.

In addition, the need for additional training for both Operations and Engineering personnel was recognized. Engineers and other appropriate personnel were trained in June 1996 by corporate Nuclear Engineering in "Regulatory Fundamentals" regarding recognition of departures from the design/licensing bases, and the associated temporary and permanent response requirements. Training on the new station procedure was also conducted in September 1996.

Currently, any station personnel can initiate an operability issue by writing a PIF. All Engineering personnel have been trained as to when to initiate PIFs. The PIF process is discussed in Action (d). The current version of the station's PIF process provides many examples of potential degraded or non-conforming conditions that would prompt non-engineering or non-operability trained personnel to initiate a PIF for those conditions. Oversight regarding operability concerns

is also provided through the PIF review process. Engineering and Operations personnel have been trained and are expected to understand the licensing/design bases of the plant and to recognize potential deviations. When in doubt during a PIF review, the Shift Manager initiates an operability review.

1.18 Plant Configuration Management Programs

In addition to the Out-of-Service program described in Appendix II, Process 20, the following additional processes are in place to assure the plant is configured and operated in accordance with the design and licensing bases. These programs include: 1) Start-Up readiness Review; 2) The Operational Safety Predictor (OSPRE); 3) Shutdown Risk Management; 4) Abandoned Equipment; 5) Operations Checklists; and 6) Discrete Component Operations, and are discussed below.

1.18.1 Start-Up Readiness Review

A procedurally mandated administrative review of outage activities is performed prior to a unit start-up following a refuel or other outages of a long duration, or as deemed necessary by the Operations Manager, to ensure that the unit is safe to start-up and operate for the reminder of the operating cycle. This review includes:

- Outstanding NTS items and other commitments
- Out-of-Services on systems required for startup
- Open Temporary Alterations
- System walkdowns results
- Open work requests
- Outstanding design changes
- Operability evaluations
- Primary Containment integrity
- Surveillances and PMTs
- others

1.18.2 Operational Safety Predictor (OSPRE)

OSPRE is a computer program (controlled by an administrative procedure) which contains a database of the Core Damage Frequency for combinations of unavailable components obtained from PRA analysis used by the station to provide risk-related information about the current or proposed plant configuration. For components included in the OSPRE database, OSPRE is used to determine, from a safety perspective, which equipment should have repair priority and which can be safely taken out-of-service for maintenance.

1.18.3 Shutdown Risk Management

The purpose of the Shutdown Safety Management Program is to ensure adequate plant protection during outage periods through pre-outage reviews of planned outages, on-going reviews of the shutdown plant conditions, and increased awareness of plant conditions during plant outages. This program is controlled by station procedures.

1.18.4 Abandoned Equipment

Equipment that is no longer required and is performing no useful function, may be permanently isolated. The abandoned equipment process is used in conjunction with the design change process described in Section 1.5. Conformance with the design and licensing bases is maintained through a detailed review which addresses impact to the Technical Specifications, UFSAR, Licensing or Regulatory requirements, procedures, drawings, and other configuration management issues. This process is controlled by a station administrative procedure.

1.18.5 Operations Checklists

Use of detailed proceduralized "Systems Checklists" are specified as a prerequisite to system operation to ensure that plant configuration is maintained. These checklists are controlled and maintained through the normal procedure process as described in Action (b).

1.18.6 Discrete Component Operations

When the operation of components for a specific activity is not covered by an existing procedure, e.g., valve lubrication in off-normal situations, the Discrete Component Operation (DCO) process is utilized. The DCO describes the specific conditions under which the component(s) will be operated and provides the actions necessary to complete the required task. Compliance with the plant's design and licensing bases is controlled through the review process which is performed by a Senior Reactor Operator. This process is controlled by a station administrative procedure.

1.19 Confidence In Processes

1.19.1 Engineering Assurance Group (EAG)

Due to issues raised by the NRC Independent Safety Inspection (Fall 1996) concerning the station's control of documents which form the plant's design bases, Dresden has committed to perform several follow-up actions. On an interim basis, Dresden has established an Engineering Assurance Group (EAG) with a charter to oversee key Engineering activities to ensure that the design bases are validated, maintained, and if necessary, reconstituted. This group, made up of senior ComEd Engineering personnel and experienced outside experts, reviews and evaluates the following:

- Design Changes for new modifications and associated calculations
- Operability Evaluations

- 10 CFR 50.59 Safety Evaluations
- Engineering evaluations performed at the request of the plant
- Evaluations for Temporary Alterations
- Special Test procedures
- Surveillance Trending, e.g., ISI, IST, MOV Testing
- PIF process performance
- Other items as requested by the Site Engineering Manager

1.19.2 Indicators Monitored and Displayed in the 10 CFR 50.54(f) Letter

DCR Drawing Backlog

This indicator measures the schedule performance for Document Change Request (DCR) Control Room drawing update/turn around time. Monitoring the DCR drawing backlog for the initiation, engineering review, incorporation, and closure assures DCR drawings are processed expediently to maintain the design performance of the unit(s).

Modification Package Disposition after Operation Authorization

This indicator measures the schedule performance for dispositioning the modification package. Dispositioned modification packages include all ECNs, Modification Approval Letters, checklists, marked up proposed changes to UFSAR, Technical Specifications, VETIP, testing and other procedures, etc. The dispositioned package contains the actual revised and approved documentation or a reference to a tracking number for the marked up and proposed revisions to various documents. Monitoring the modification package disposition after Operation Authorization ensures that documents are updated and the plant's safety related design is maintained.

Temporary Alterations

This indicator monitors Temporary Alterations to the approved design configuration of a structure, system, or component (SSC). The primary purpose is to provide assurance that a Temporary Alteration made to plant equipment is accounted for to maintain plant safety and reliability.

Backlog of Technical Manuals

Monitoring the Backlog of Technical Manuals is an important measure that ensures the program has current and complete vendor data. An effective manual program ensures the plant equipment is operated and maintained to conform to the original design.

Operator Workarounds

This indicator is used to monitor the plant design deficiencies that could affect the original performance and design basis of the plant. It includes operator compensatory actions

(workarounds) during plant transients that involve equipment deficiencies, including design deficiencies and procedures.

Engineering Requests

This indicator monitors the number of several engineering request (ER) types. It displays the number of open unplanned engineering work load that could effect the plant design basis.

2.0 Action (b) Rationale for Concluding that Design Bases Requirements are Translated into Operating, Maintenance, and Testing Procedures.

2.1 Introduction

Dresden Station implements a comprehensive procedure preparation and revision process, in accordance with applicable licensing and Quality Assurance requirements. This process provides assurance that appropriate design bases requirements are translated into operating (normal, abnormal, and annunciator response procedures), maintenance, and testing procedures. In those cases where procedures have been found not to be consistent with design bases, action was taken to resolve the deficiency. In some cases, this has resulted in the need to modify the procedure development process and perform select procedure upgrades, as described below, for sustained performance improvement.

Dresden's rationale for concluding that reasonable assurance exists that design basis requirements are translated into operating, maintenance, and testing procedures is based on the following:

- 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. When the plant was licensed, Dresden demonstrated to the NRC that these original procedures provided assurance that design bases requirements had been translated into operating, maintenance, and testing procedures.
- Subsequent to startup, new procedures and procedure revisions at Dresden have been prepared in accordance with applicable Station Administrative Procedures that encompass Quality Assurance and license requirements. These Station Administrative procedures incorporate a number of reviews (checks and balances) which are intended to assure that applicable design basis requirements are considered prior to the approval and use of each procedure revision or new procedure. A standard Procedure Writer's Guide is followed for procedure development. Use of this guide also helps ensure consistency in preparing and maintaining high quality procedures.
- Operating, maintenance, and testing procedures have been implemented at the Dresden plant for many years. The resulting consistency between expected and actual station responses is one measure that design basis information has been translated into these procedures accurately.
- Dresden has completed a major procedure upgrade project to standardize procedure format and content. This effort required additional design reviews and verification of established procedures, and further ensured consistency with design bases requirements.
- Certain configuration documentation improvement programs and assessments implemented at Dresden included conformancy checks against operating and/or maintenance and testing procedures.

- The number and types of procedure deficiencies, as obtained from a review of recent Performance Improvement Form (PIF) suggests no significant deviations from design bases in the plant's configuration, and no adverse trends in this area. Although numerous procedural discrepancies were identified, none was considered to be of major significance.
- Procedure change requests are addressed promptly when inconsistencies with design bases are identified.

2.2 Consistency of Original Station Procedures with Plant Design Bases

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

Key elements of the procedure review process and qualification requirements of personnel who perform these reviews were established through original plant Technical Specifications. These reviews include one or more of the following: Technical Review; Onsite Review; Verification; Plant Operations Review Committee (PORC); and Plant Manager review and approval. This multi-level/multi-discipline review by qualified personnel helps ensure procedure consistency with design bases. (see Section 3.3)

2.3 Procedure Preparation and Revision Process

The procedure preparation and revision process at Dresden requires multiple plant reviews and checks to confirm that design requirements and design bases are correctly translated into procedures. Procedure preparers and reviewers have access to needed design basis source documents, e.g., UFSAR, Technical Specifications, drawings. In addition, a Procedure Writer's Guide is used which outlines basic procedure content requirements and format, and defines the method for flagging commitments contained in procedures for future control and compliance.

Procedures are prepared by experienced individuals and are reviewed by qualified personnel in a multi-level/multi-discipline review process who are knowledgeable of design bases information location and use. This multifaceted review is a key element of the procedure preparation process. It includes (as appropriate) Technical Review, Onsite Review and Verification, and Human Factors Review.

The procedure preparation and revision process includes several of the following steps which provide the checks and balances that help assure that design bases information is accurately translated into operating, maintenance, and testing procedures:

- 10 CFR 50.59 Screening and Safety Evaluation
- Technical Review

- Onsite Review
- PORC
- Station Manager Review
- Verification
- Commitment Preservation
- External Information

At a minimum, new procedures and procedure changes are required to undergo a technical review and a 10 CFR 50.59 Screening prior to approval. Additionally, new procedures require verification. For procedures which have a possible safety impact and/or as required by NRC regulations, additional reviews are performed by Onsite Review, PORC, and the Station Manager.

Additionally, the design control process has been revised to ensure that station procedures are carefully reviewed to identify those which are impacted by a design change. Revisions to those procedures are tracked to completion through the design change process.

2.3.1 10 CFR 50.59 Screening and Safety Evaluation

A 10 CFR 50.59 Screening is performed on new or revised operating, maintenance, and testing procedures at Dresden to determine whether the proposed change could involve an Unreviewed Safety Question or a change to the Technical Specifications. This screening provides a check of the procedure change against license requirements and the design basis contained therein. Personnel who perform this screening must meet the qualification requirements specified in ANSI N18.1 for minimum education, training, and power plant experience required to function in this role. These requirements have been found sufficient to assure that procedure preparers and reviewers have the necessary knowledge of design bases information. Depending on the results of the screening, a safety evaluation may be required.

2.3.2 Technical Review

Technical reviews are performed on new or revised operating, maintenance, and testing procedures at Dresden to confirm technical adequacy and compatibility with existing station design and operation. Technical reviews are performed by personnel knowledgeable in the subject matter and who meet the applicable experience requirements specified in station procedures. More than one technical reviewer may be assigned; however, at least one reviewer is a member of the department for which the procedure is intended. Procedural guidance followed during technical reviews mandate consideration of several factors, including: 1) review of applicable station drawings; 2) determination whether the procedure or revision addresses lessons learned and Station commitments; and 3) impacts on systems, other procedures, other programs (Environmental Qualification, Inservice Inspection/Inservice Testing , etc.), other departments, personnel safety, commitments, safety-related equipment, and station or control room operations. By applying the guidelines, reviewers apply design bases information to the review process.

2.3.3 Onsite Review

Administrative procedures required per NRC Regulatory Guide 1.33, Revision 2 (and other selected procedures) receive a review by the Onsite Review and Investigative Function. Onsite Review is conducted in accordance with procedural guidance that include attributes for key attention such as 1) determinations of fulfillment of Technical Specification, UFSAR, and station requirements and commitments; 2) safety issues; 3) review of 10 CFR 50.59 Safety Evaluations for Technical Specification and FSAR application; 4) procedural compliance; and 5) radiological concerns. Reviewers must meet certain discipline requirements established by station procedures, per ANSI N18.1.

All procedures which require an Onsite Review are also screened by a designated knowledgeable individual for potential safety impact on plant operation. If a possible safety impact is identified, the procedure is reviewed by Plant Operations Review Committee (PORC) which consists of senior station management (Operations, Maintenance, Radiation Protection, Chemistry, Systems Engineering, and other departments) who provide multiple perspectives on the adequacy of the procedure under review.

2.3.4 Station Manager Review

Those Dresden procedures which require Onsite Review are also required to be reviewed and approved by the Station Manager. Per station procedure, the Station Manager independently reviews the findings and recommendations developed by the Onsite Review, and PORC review as applicable. Station Manager review includes a determination of the following:

- Whether appropriate participants were selected for the Onsite Review;
- Whether the review was in sufficient depth;
- The reasons for dissenting opinions;
- The reasonableness of the findings and recommendations; and
- Whether or not a PORC was required.

2.3.5 Verification

New Dresden procedures require verification by an independent, knowledgeable individual in accordance with detailed procedural guidance. Procedure revisions may also require verification as deemed necessary by the appropriate plant Supervisor. Verification consists of reviewing the prepared or revised procedure to determine whether it:

- Conforms to the Writer's Guide;
- Meets its stated purpose;
- Provides acceptance criteria or alert/action ranges in test/calibration procedures;
- Includes adequate steps and information to perform the intended function;
- Includes equipment numbers that are identical to labels in the field;
- Specifies required equipment, test equipment, personnel support, and prerequisites;
- Includes the appropriate sequencing of actions; and

- Is in agreement with expected plant/equipment response.

Emergency Operating Procedures (EOPs) are submitted to a validation process. This requires utilizing the station's Simulator to the maximum extent possible to evaluate and exercise the full scope of the procedure under a wide range of conditions, including multiple failures and events more severe and less severe than those compromising the plant's design bases.

2.3.6 Commitment Preservation

A key element of the Writer's Guide is the requirement to specifically identify, by annotation, 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 adequate review and technical justification.

An additional tool used to assist in maintaining design bases configuration at Dresden is a "Technical Specification Matrix" which is a matrix of Technical Specification Surveillance Requirements and Limiting Conditions for Operator Driven requirements and the applicable implementing procedure.

2.3.7 External Information

In addition to the specific controls within the procedure process, some of the more pertinent of Dresden Administrative procedures, which interface with the procedure process, provide additional procedure controls. These include:

- Plant Design Changes
- Setpoint Change Control
- Temporary System Alterations
- Alternate Replacement Program
- Vendor Equipment Technical Information Program (VETIP).

The procedures controlling these processes ensure that applicable industry information and vendor equipment technical information is reviewed and incorporated into applicable Dresden plant procedures, and that procedures affected by modifications are identified and revised accordingly.

The procedure development and revision process, required reviews, reviewer qualification requirements, and procedure content as specified by the Writer's Guide all help ensure that design basis requirements are incorporated into plant procedures.

2.4 Experience with Procedures

Procedures have been implemented at Dresden for many years and have proven their effectiveness through experience. Some examples of major plant events which confirm the adequacy of procedures include startup, shutdown, refueling operation, and simulated emergency and casualty

operation conducted in the Simulator. Additionally, the plant has experienced the following unplanned events:

- Loss of Feedwater Flow
- Loss of Offsite Power
- Loss of Feedwater Heating
- Turbine Trip with By-Pass
- Load Reject with By-Pass
- Main Transformer Fire

Procedures have proven to provide adequate guidance to the operators. Where improvements were required, applicable procedure revisions were made.

2.5 Procedure Upgrade Program

In response to the NRC's identification of numerous errors in Dresden's procedures a commitment was made to review and upgrade all station procedures to ensure consistency in format and content. This major upgrade effort was initiated in April 1988 and completed in April 1993. A team of Procedure Writers with varied backgrounds, e.g., mechanical, electrical, operations, was assembled to accomplish the task. A Procedure Writer's Guide was developed, based on industry guidelines, to provide guidance for ensuring consistency in preparing and maintaining high quality procedures. The Upgrade Program required all station procedures to be reviewed against the FSAR, Technical Specifications, Vendor Manuals, Commitment Tracking, the As-Built conditions of the plant, and a Human Factors Review. This effort helped ensure procedure conformance with design bases. This effort included writing numerous maintenance procedures. Testing procedures were also strengthened during this process.

2.6 Assessment and Audit Results

A number of documentation improvement programs have been conducted at Dresden over the years. These initiatives confirmed on a sampling bases that design information has been translated accurately into operating, maintenance and testing procedures. Where deficiencies have been noted, they have been corrected.

2.6.1 UFSAR Review Program

In 1996, 100% of the UFSAR was reviewed against Dresden Operating Procedures (DOPs) and Dresden Operating Surveillances (DOSs) to ensure that UFSAR requirements were being adequately translated into station operating practices. Approximately 98 packages, organized by system and UFSAR section, were prepared for review. The reviewers were familiar with the particular system and procedure/surveillance in question. They evaluated the procedure/surveillance against related UFSAR issues and provided comments and suggestions for updating the UFSAR and/or procedures.

Additionally, four systems (Core Spray, Low Pressure Coolant Injection/Component Cooling Service Water, High Pressure Coolant Injection, and 125VDC) were selected for a more in-depth review. In addition to reviewing the related DOPs and DOSs, a review of the Technical Specifications, Dresden General Abnormal (DGA) procedures and Dresden Annunciator (DAN) procedures was performed.

Based on the above two review efforts, a total of 389 potential concerns were identified and evaluated:

- Potential Operability Concerns 0 Findings
- UFSAR/Procedure Inconsistencies 13 Findings
- Potential UFSAR Updates 348 Findings
- Potential Procedure Updates 28 Findings

Procedure revisions have been completed. Potential UFSAR concerns were evaluated and appropriately dispositioned. Those determined to impact the UFSAR were addressed in one of the forty one UFSAR change packages which were initiated. As of January 1997, 37 have been completed and 4 are in the preparation and review process.

As a result of this review, Dresden has reasonable confidence that operating procedures are substantially consistent with the plant's design and licensing bases.

2.6.2 Primary Containment Assessment Program

In January 1996, the NRC approved the use of Option B to 10 CFR 50, Appendix J for Dresden Station. Primary Containment Leakage performance was evaluated and it was determined that both Unit 2 and Unit 3 Integrated Leak Rate Tests (ILRTs) could be placed on a 10 year frequency. All associated station procedures were reviewed for conformance. Eleven Local Leak Rate Tests/ILRT procedures were revised to implement the required changes. The UFSAR was revised to eliminate any conflicts with the Option B performance based Technical Specifications.

2.6.3 Upgraded Technical Specification Review

The Technical Specification Independent Review Group (TSIRG) was established in 1994 to perform a comprehensive, line-by-line review of Dresden's proposed Upgraded Technical Specifications. The objective was to identify concerns that could result in non-compliance with the Upgraded Technical Specifications upon implementation. Identification of such concerns would help prevent failures to comply with Technical Specifications because of inappropriate implementing procedures and/or misinterpretations between the Upgraded Technical Specifications and their implementing procedures.

ComEd and contractor personnel with previous licensing, operating, and Technical Specification experience were enlisted to perform the review. Through a series of questions the Upgraded Technical Specifications, Sections (2, 3, 4, 5, and 6) were compared line-by-line to key plant documents (UFSAR, Station procedures, Operability Assessments, and Technical Specification

Interpretations) to identify deficiencies (errors, omissions, conflicts between documents, and unclear or vague language) which might lead to failure to comply with the Upgraded Technical Specifications.

A series of questions specific to each document type was used to focus the review. The scope of the project required the review be narrowed by a set of assumptions which would still allow adequate inspection of plant documents to identify problem areas. The process used for the review included computer searches for procedures and UFSAR information, and use of the Procedure-Writing Group's Technical Specification Cross-Reference Matrix as a starting point.

Three issues were identified which consisted of problems with compliance with the current Technical Specifications. These issues were resolved.

Approximately 1750 other potential Technical Specification and procedure comments were generated, including administrative items, technical questions, and the identification of needed procedure revisions. These items were reviewed, appropriately dispositioned and either eliminated or resolved prior to Technical Specification Upgrade implementation.

2.6.4 System, Valve Line-up Checklist Walkdowns

In March 1996, Dresden Station management determined that a thorough review of Unit 2 system checklists was needed, and subsequently decided to perform a similar review on Unit 3 checklists. A multi-discipline team of Dresden personnel and third party industry peers was established to conduct a comprehensive walkdown to ensure that the Dresden Operating Procedures (DOP) checklists are accurate and complete. In addition, Engineering and Operations performed a review of the UFSAR and Technical Specification requirements concerning locked valves. Valves which are required to be locked in accordance with the UFSAR and Technical Specification were verified prior to startup, and the appropriate revisions to the procedures were completed. The results of the system checklist walkdowns have given Dresden Station confidence that, relative to valve positions, the plant operating procedures are configured in accordance with its design basis. Procedures were revised to enhance controls to assure that future problems would not occur.

2.6.5 SQV Audit of Nuclear Engineering Procedures (NEP) Implementation

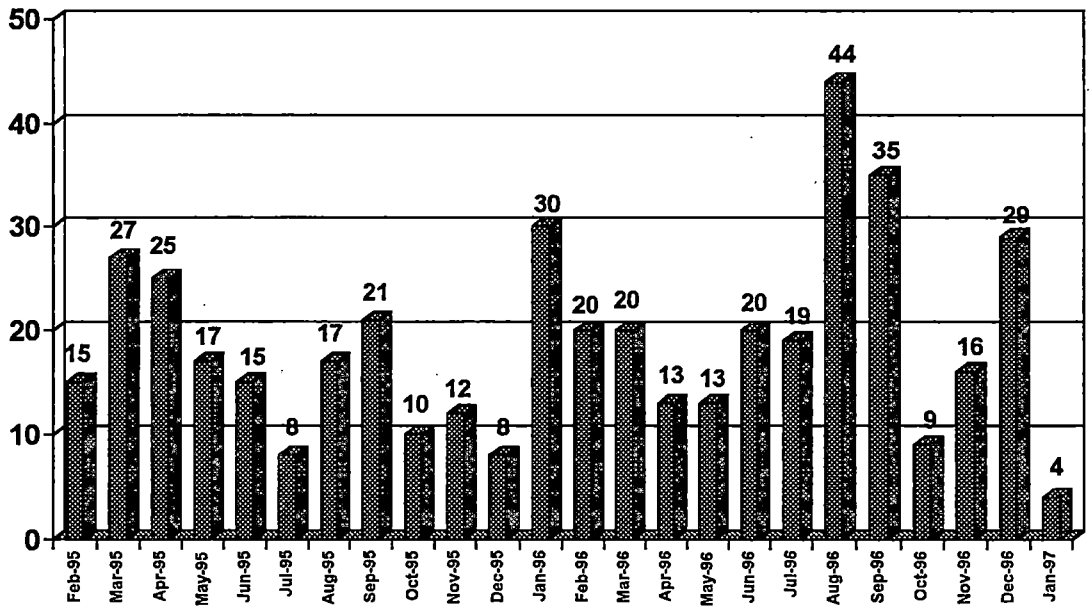
NEPs implement Dresden's design control processes. In December of 1996, SQV performed an audit of the adequacy of Dresden's implementation of and compliance with the NEPs. Two Level III findings were identified as a result of this audit, neither of which would impact the design bases of the plant.

- An inadequate training matrix for training to new/revised NEPs resulted in lapses of compliance with the NEPs. An NTS item was initiated to track correction of this item and a SQV follow-up is scheduled for February 1997.
- Several administrative errors noted in the completion of 10 CFR 50.59 forms. An NTS item was initiated to track correction of this item and a SQV follow-up is scheduled for February 1997.

2.7 PIF Trends and Data Analysis

Procedure deficiencies are identified through Performance Improvement Forms (PIFs). PIFs related to "Procedure Deficiencies" receive a specific tracking code which allows them to be tracked and trended. Approximately 500 PIFs are initiated each month at Dresden. The number and types of procedure deficiencies, as obtained from a review of recent PIFs suggests no significant deviations from design bases in the plant's configuration, and no adverse trends in this area. Approximately 4% of the total PIFs issued were related to procedure deficiencies. Few procedure discrepancies were identified, and none were considered to be of major significance.

Procedure Adequacy-related PIFs
February 1995 to Present



2.8 Improved Procedure Revision Process

In February 1996, Dresden implemented a simplified procedure revision process allowing procedure revisions to occur in a more expedient manner. Unnecessary Onsite Reviews were eliminated and clerical work associated with the change was put at the back end of the process. The Writer's Guide was streamlined accordingly. The process now encourages individuals to correct procedural discrepancies as they are discovered, thus ensuring that procedures are maintained current.

2.9 Conclusion

Based on the multilevel reviews and SQV oversight in the procedure update and revision process, established reasonable assurance exists that design bases requirements are translated into the operating, maintenance, and testing procedures at Dresden Station. In addition, procedures have been the subject of ongoing review throughout the operating life of the plant. When need for improvement is identified, procedural upgrades have been made.

3.0 Action (c) Rationale for Concluding that Structure, System, and Component Configuration and Performance are Consistent with the Design Bases

3.1 Introduction

ComEd concludes that for Dresden Station there is reasonable assurance that the configuration and performance of its structures, systems, and components (SSCs) are in substantial compliance with the design bases. The basis for this conclusion can be summarized as follows: When Dresden was licensed to operate by the NRC, the license was supported by information provided by ComEd demonstrating that Dresden's SSCs were constructed in accordance with the plant's design bases. Since then, Dresden 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 activities have been conducted in accordance with processes and procedures designed to preserve the configuration and performance of SSCs consistent 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, generate a substantial body of experience that demonstrates conformance of the SSCs with their design bases. Also, a large body of data about SSC configurations has been developed over the years as various SSCs are reviewed for modification or maintenance; subjected to special verification and improvement initiatives; 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.

ComEd recognizes that several recent ComEd and NRC assessment activities have identified weaknesses in configuration management processes at Dresden Station. In addition, there has been an historic weakness at the Station related to the timeliness of identifying and resolving issues, as evidenced by problem backlogs or long-standing programs. ComEd has recently performed a series of reviews (described in Section 3.7.2.2) to reverify that current plant conditions are safe and support continued operation, and has confidence for Dresden that the backlogs related to design bases does not raise operability issues or issues involving conformance with the design bases. These reviews provide confidence that the design bases and engineering controls support safe operation. In addition, ComEd has previously committed to several significant actions to upgrade the quality and access to design information at all six nuclear stations. These actions will go forward as Dresden Station moves to improve performance in this area.

The response to Action (c) is structured as follows:

Section 3.2 Initial Determination that SSCs were consistent with Design Bases;
Section 3.3 Preservation of the Design Bases;
Section 3.4 Ongoing Verification Activities;
Section 3.5 Operating Experience;
Section 3.6 Special Verifications and Improvement Initiatives;
Section 3.7 Audits, Inspections, and Configuration/Performance Assessments; and
Section 3.8 Conclusion.

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

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

Dresden Station's design bases were established prior to issuance of many of the industry and regulatory standards applied to newer plants. In particular, Dresden was constructed before the NRC codified the General Design Criteria (GDC). The GDC were issued in a draft form in July 1967 to obtain comments from the industry and a response on whether the plant, as designed and being constructed, could meet the criteria. Compliance with the intent of the AEC's proposed GDC is addressed in Section 3.1.1 of the Dresden UFSAR. After the FSAR was filed, the AEC asked ComEd to demonstrate that Dresden Station design complied with each of the final General Design Criteria published as Appendix A to 10 CFR 50 in July 1971. The response was supplied as an informative comparison and is presented in Section 3.1.2 of the UFSAR. Section 3.1.2 of the Dresden UFSAR states:

> This section contains an evaluation of the design basis of the Dresden Nuclear Power Station Unit 2 as measured against the NRC General Design Criteria (GDC) for Nuclear Power Plants, Appendix A of 10 CFR 50, effective May 21, 1971, and subsequently amended July 7, 1971. The General Design Criteria, which are divided into six groups, are intended to establish minimum requirement(s) for the design of nuclear power plants...
>
> ... Based on the material contained in this application, ComEd concluded that Dresden Station Unit 2 satisfies and is in compliance with the intent of the General Design Criteria. This evaluation was performed specifically for Unit 2 and may not fully apply to Unit 3; however, the high degree of similarity between the design of Units 2 and 3 indicates that Unit 3 also conforms to the intent of the General Design Criteria. Additional design information pertinent to each criterion is addressed in section of the UFSAR describing various systems and structures.

Chapter 3 of the UFSAR describes the design of the Dresden structures, components, equipment and systems and their conformance with the General Design Criteria.

In the late 1970s and into the early 1980s, the NRC and ComEd conducted an extensive evaluation of Dresden Unit 2's conformance to the GDC. This evaluation was called the Systematic Evaluation Program (SEP). In most cases, Dresden Unit 2 was found to meet the applicable GDC. In some cases, literal compliance was not achieved; however, the intent was met, and equivalent protection to the general public was achieved. In a very few cases, design changes were implemented to bring the plant into effective conformance with applicable GDC. When Dresden Unit 2's operating license was converted from a "provisional operating license" to a "full term operating license," compliance with SEP findings was verified.

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

Processes were established to maintain Dresden's configuration and performance in conformance with its design and licensing bases. 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) and Appendix II of this response. As was discussed in Action (a) and Appendix II, 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.

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 (as identified in procedures, drawings and databases) are obtained. Some of the routine performance monitoring activities include plant walkdowns, surveillance testing, post maintenance testing, post modification testing, and implementation of the Maintenance Rule. Each of these activities is described in more detail below.

3.4.1 Plant Walkdowns

The configuration of SSCs is maintained in part by plant personnel during their performance of regular duties. Operating procedures require plant rounds to be performed on a regular basis, during which Operating Department personnel record parameters which indicate whether SSCs are operating in accordance with procedural requirements. Operating parameters such as pressures, flows, temperatures, vibration, and oil levels are routinely monitored. SSC problems are identified during these walkdowns and are documented in accordance with Dresden's corrective action program. Issues potentially impacting equipment operability are brought to the attention of plant management and processed in accordance with plant procedures for assessment.

The System Managers (System Engineers) also conduct formal, documented walkdowns of accessible selected SSCs on a periodic basis. The Plant Engineering Handbook defines a system walkdown as:

> the focused effort, physical and mental, that Plant Engineering periodically places on a system or portion of a system consisting of a visual inspection, system performance review, and proactive reliability review to determine current health of the system, degradation from last walkdown, and identification of potential problems.

The System Engineers have received formal training and qualification in system walkdown techniques. The walkdowns review general system status, the condition of electrical panels and mechanical components, general material condition items, safety hazards in the system area, housekeeping items, radiological items associated with the system, and various logs and round sheets that provide useful information about the status of a system. The walkdown requires discussions with operations personnel to obtain their insight into the operation of the system. This walkdown program partially implements management expectations that System Engineers constantly monitor their system's performance, thus helping to provide assurance that systems are being operated in accordance with their design and licensing bases.

Operator and System Manager walkdowns are ongoing processes to continually compare system configuration and status against the station's design and licensing bases, thus providing opportunities to discover discrepancies in those bases. The walkdown programs, as well as other less formal programs, help provide assurance that the station is configured, maintained and operated in conformance with its design and licensing bases.

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 assure reliable performance upon demand throughout the plant lifetime.

Periodic surveillance testing is performed in accordance with Technical Specification requirements and Inservice Testing Program requirements. The testing procedures verify that critical system performance parameters are satisfied during system operation. Testing discrepancies require evaluation for operability. In addition, conditions adversely impacting system operability are evaluated for root cause and corrective action determination via the station corrective action Integrated Reporting Program (IRP) process (described in Action (d)).

3.4.2.1 Inservice Inspection (ISI) Program

One aspect of surveillance testing conducted at Dresden is the Inservice Inspection (ISI) Program, which implements the station requirements for non-destructive examination of Class 1, 2, and 3 pressure retaining components and their supports. The ISI program covers such items as welds,

vessels, pumps, valves, supports, and snubbers. The program was developed in accordance with the requirements delineated in the July 31, 1991 issue of 10 CFR 50.55(a) and the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section XI. The components selected for examination, the frequency of examinations, and the examination techniques employed are as stipulated by ASME Section XI, as augmented by 10 CFR 50.55(a). The ISI program also governs the repair and replacement of Class 1, 2, and 3 pressure retaining components and their supports at Dresden Station. The upper tier document governing the ISI Program is 10 CFR 50.55(a), which provides the rules for development and implementation of ISI Programs, identifies the year of ASME Section XI that is to be followed in its implementation, and provides augmented requirements to be implemented above and beyond those of the ASME Code.

Dresden is in its third 10 year ISI inspection plan interval. The key plan features include: Plan Description, Relief Requests, Technical Approach and Positions, and Component Summary Tables. This plan is a controlled document that was submitted to and approved by the NRC in accordance with 10 CFR 50.55(a). Dresden also plans to develop an ISI program for Class MC (metal containment) components in accordance with the requirements delineated in the September 9, 1996 revision of 10 CFR 50.55(a).

Dresden's ISI program provides the opportunity for experienced engineering personnel to compare the physical plant configuration against information contained in station drawings, ISI plan documentation, equipment databases and other design and licensing basis information. The fact that Dresden has completed two 10 year interval plans has afforded an opportunity to correct discrepancies between the physical plant and information contained in engineering data. It also assures that degraded components identified are restored to an acceptable condition, thus assuring that the station is operated in conformance with design and licensing bases. The Section XI Repair and Replacement program helps assure that, as maintenance and modifications are performed, the station's design and licensing basis is maintained.

As part of Dresden's program to constantly improve station programs, a self-assessment of the RPV In-Vessel Examination Program was performed in July 1996, using an expert from outside ComEd. The scope of the self-assessment consisted of a review of all aspects of the program, including compliance with existing Codes, augmented requirements, and NSSS recommendations, along with implementation of existing and draft owners group recommendations. No areas of non-compliance were identified.

3.4.2.2 Inservice Test (IST) Program

Another aspect of surveillance testing at Dresden is Inservice Testing (IST), which is used to verify the operability of safety-related pumps and valves and measure component degradation to provide assurance of continued operability. The current third 10 year inspection interval for both Units is from March 1, 1992 through February 28, 2002. ASME Code-Class and Non-Code class pumps and valves that may function to mitigate the consequences of an accident, to bring the plant to the safe shutdown condition, or to maintain the plant in the safe shutdown condition, are included in the IST program and are tested commensurate with the importance of the safety

functions to be performed. Augmented tests may also be imposed by the NRC or voluntarily included in the IST Program as good engineering practice.

On June 28, 1996, a self-assessment of the IST program was performed to verify that program requirements are defined and implemented in accordance with ASME Codes, standards and regulatory requirements. The assessment focused on safety and relief valve testing, check valve inspections, valve seat leakage testing, and test instrumentation. During this self-assessment, several issues were identified. The overall conclusion of the self-assessment was that a full program assessment was required and that an IST and Augmented Testing Program Basis document was needed. The 1996 NRC Independent Safety Inspection identified concerns similar to those discovered by the June 1996 self-assessment, as well additional issues such as, inconsistencies between Design Bases Document and IST valve stroke time values. Dresden believes the current state of the IST program is robust and acceptable with minor problems. This is based on reviews of the results of the self-assessment, the progress of the "Basis Review" to date, and the extensive review by the 1996 NRC ISI team with respect to current regulatory requirements.

The IST Program Basis document will contain component safety functions, testing requirements, and the basis for inclusion or exclusion of components from the IST or Augmented Testing Programs. A detailed project plan was written and approved to identify the methodology and scope for performance of the assessment and development of the Testing Program Basis Document. This review, which began in August 1996, is being conducted on a system-by-system basis, starting with the ECCS systems; the project is scheduled to be completed in 1997. At the conclusion of this project, a revised IST Plan will be submitted to the NRC.

The IST program is another example of intrusive activities which afford engineering personnel the opportunity to examine the physical plant and compare it to design and licensing basis information. The results of the IST program help provide assurance that the station is being operated in accordance with its design and licensing basis. The self-assessment will provide another opportunity to confirm that the plant is configured and operated in conformance with its design and licensing bases.

3.4.2.3 Local Leak Rate Test (LLRT) and Integrated Leak Rate Test (ILRT) Programs

The purpose of the LLRT and ILRT programs is to determine leakage through primary containment barriers in order to assure that primary containment integrity is maintained and Technical Specification requirements are met. Leakage results are used to evaluate and trend component performance, establish surveillance intervals, and report leakage findings. Compliance with these requirements helps assure that the station is operated within applicable design and licensing basis criteria.

The LLRT and ILRT programs have been revised in 1996 to implement Option B to 10 CFR 50, Appendix J. A self-assessment to ensure that reverse direction local leak rate testing is as conservative (or more conservative) than accident direction local leak rate testing was performed.

In September 1995, of the 100 valves previously tested in the reverse direction on Unit 2, 74 were tested in the accident direction during D2R14 in 1995 and the remaining 26 were qualified to be equivalent or more conservative than testing in the accident direction. Of the 99 valves on Unit 3 previously tested in the reverse direction, 94 were qualified to be equivalent (or more conservative) than testing in the accident direction and five were retested in the accident direction. All volumes passed their LLRTs under the new testing methodology.

This Dresden 10 CFR 50, Appendix J program provides assurance that the Primary Containment Isolation Valves will meet their design and licensing basis functions.

3.4.3 Post Maintenance Testing and Modification Testing

The plant modification process requires Design Engineering to identify appropriate Construction, Modification 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. Prior to integrating with the plant systems, Construction Tests are performed to ensure that installation work is performed correctly and in accordance with the governing codes and standards. Modification Testing ensures that the plant modification 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 either in the work package (for basic testing) or by special tests (for more complex tests). This testing provides added assurance that the modification complies with applicable codes and standards and is consistent with the design bases.

The Post Modification Testing (PMT) process has recently been enhanced as a result of ComEd and third-party assessments. For example, a PMT assessment was conducted in October 1996 to determine whether the modification, exempt change, and temporary alteration processes provide adequate guidance for the performance of testing, and whether the guidance has been adequately followed in a sampling of design changes. A total of fourteen design changes were selected for review. The self-assessment identified a weakness in the modification testing process in the area of engineering management's communication of modification testing expectations. This was evidenced by a lack of attention-to-detail, inconsistencies, and omissions relative to modification testing in several of the plant design change documentation packages reviewed. These findings were consistent with issues identified in the 1996 NRC Engineering and Technical Support (E&TS) inspection (see Section 3.7.1.6). Appropriate corrective action are being implemented.

The plant work control process at Dresden described in Action (a) of this response includes requirements for review of all work packages prior to issue for work, and for specification of any Post Maintenance Test(s) required to maintain the plant in accordance with the design and licensing bases. This process ensures that SSC performance is maintained in accordance with design bases requirements following plant maintenance work.

In October 1996, a Post Maintenance Test group was also established to specify Post Maintenance Test requirements. The 1996 NRC Independent Safety Inspection team identified that there was overlapping and ill-defined responsibilities for the new group. Additionally,

Performance Improvement Forms (PIFs) were not always being generated in response to identified problems. Subsequent to the inspection, appropriate administrative procedures have been revised and training held to address the issue of delineation of responsibilities. Engineer's use of PIFs is discussed in Action (e), Section 5.7.

Despite these weaknesses, it was concluded that testing performed under the various types of design changes complies with the respective procedural requirements in effect at the time of the plant design change was issued. The design change processes related to testing are being followed in an acceptable manner, providing assurance that the station's design bases are being preserved by acceptable modification testing.

3.4.4 10 CFR 50.65 Maintenance Rule Implementation

On July 10, 1991, the NRC published 10 CFR 50.65, which requires commercial nuclear power plant licensees to monitor the effectiveness of maintenance for specific plant equipment. In the supplementary information published with the new regulation, the NRC stated that it "believes that the effectiveness of maintenance must be assessed on an ongoing basis in a manner which ensures that the desired result, reasonable assurance that key SSCs are capable of performing their intended function, is consistently achieved." Though a recent addition to ongoing monitoring activities at Dresden, implementation of the Maintenance Rule provides added assurance that SSC performance is consistent with the identified operating parameters. Routine monitoring and assessment of the performance of SSCs that contribute most significantly to plant safety provides added assurance that these SSCs function when and as required.

The Maintenance Rule implementation and compliance program at Dresden uses the guidelines and requirements specified in NRC Regulatory Guide 1.160, NUMARC 93-01 and 93-02, the ComEd Guidelines for the Maintenance Rule Implementation, and other documents. Any deviations from NUMARC 93-01 guidelines are identified in the Maintenance Rule Implementation Procedure.

During the 1996 NRC Independent Safety Inspection, Dresden Station's Maintenance Rule program was assessed and generally considered to be satisfactory. Compliance with the Maintenance Rule requirements provide additional assurance that the plant is configured and operated in accordance with identified operations parameters.

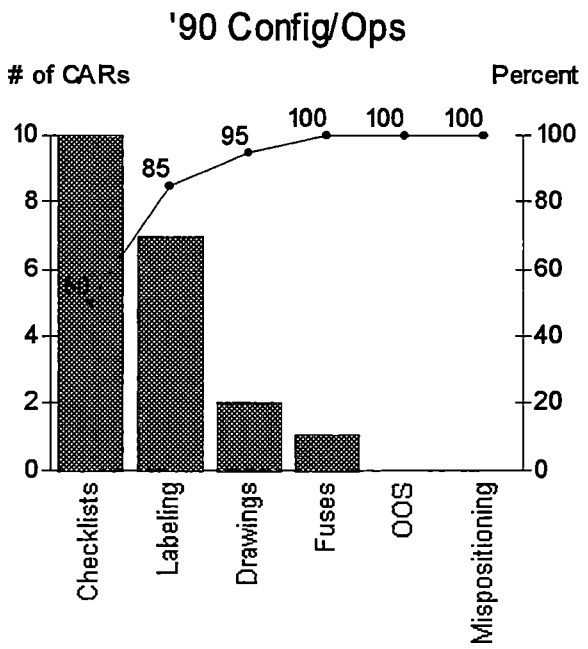
3.4.5 Site Quality Verification (SQV) Monitoring

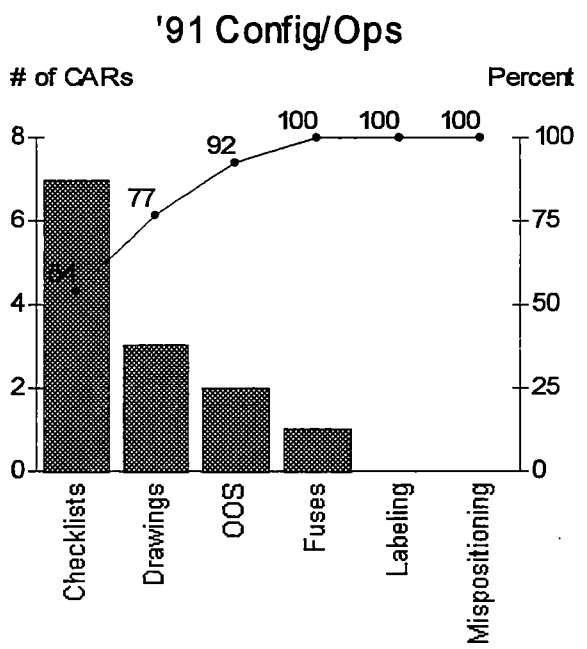
SQV performs routine and special audits of various station activities at Dresden. Findings of non-conformance are documented in Corrective Action Records (CARs). These are closed-out in the normal course, with corrective actions to the plant configuration or the documentation as necessary. This is part of an ongoing process to maintain and restore plant configuration.

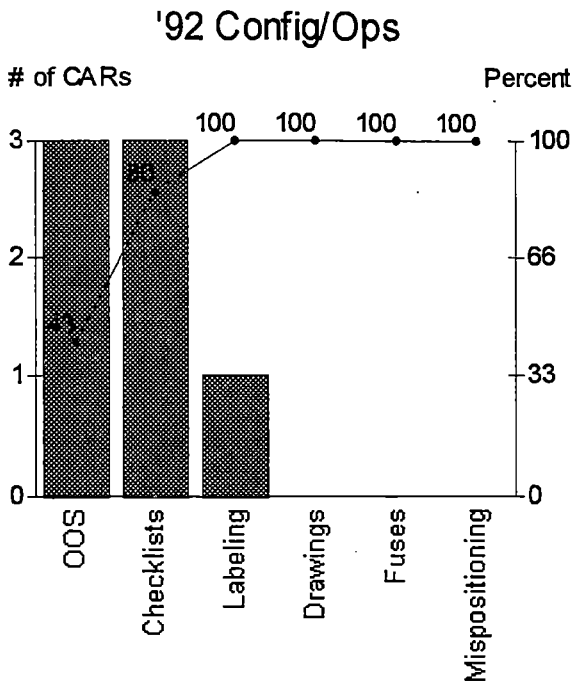
Applicable management initiatives encourage the identification and correction of off-normal conditions, including deviations from design bases. An example of this is the November 1996 stop-work-order by the Dresden SQV organization when it became apparent that parts

evaluations for balance-of-plant replacement parts were not adequately preserving the plant's design bases. A "Level 2" investigation team was formed to address these issues. A significantly strengthened process has been created to address the issues of NSR parts usage and evaluation.

In preparing for the response to Action (c), a search of the CAR database was performed for issues related to "SSC Configuration and Performance are Consistent with the Design Bases." The results are presented below. As can be seen, the predominant issues change from year to year as the station responds to varying issues. Issues sometimes reappear, either due to a dip in performance or to higher expectations of performance. Problems associated with procedure checklist fidelity and drawing errors were recurrent issues. Labeling also appeared as labeling expectations were raised. Action (d), Section 4.8.1 has additional discussion on SQV activities and effectiveness.



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Checklists
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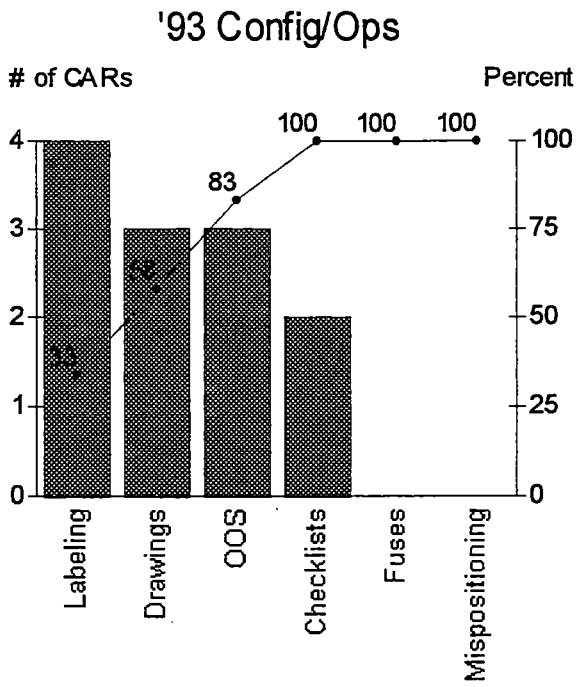
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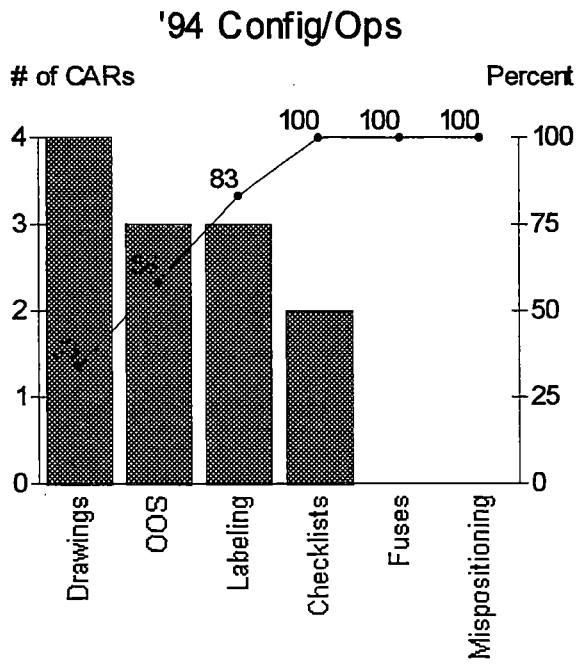
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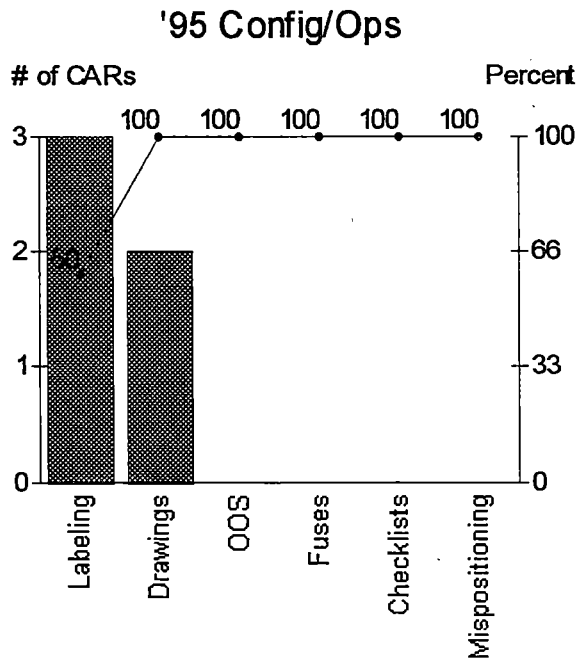
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Checklists
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3.4.6 Other Routine Third Party Assessments

In addition to the assessments discussed above, other third party assessments occur on a routine basis. These include: Authorized Nuclear Inspector (ASME Code); Hartford Steam Boiler Inspections (ASME Code); Nuclear Mutual Limited (Fire Protection); and others. Although compliance with design and licensing bases is not the primary focus of these inspection efforts, significant examination of design basis conformance occurs, especially conformance to the Fire Protection and ASME codes.

3.5 Operating Experience

During the last twenty-five years, performance of the plant provides additional verification that SSC configuration and performance are consistent with the design bases. Dresden's successful response to past transients provides some assurance that the configuration and performance of key SSCs has been maintained.

Some examples of major plant events which confirm the adequacy of plant design and configuration include startup, shutdown, refueling operation, simulated emergency and casualty operation conducted in the Simulator. Additionally, the plant's response to certain unplanned events indicated that Dresden's systems responded appropriately. These events included:

- Loss of Feedwater Flow (High Pressure Coolant Injection);
- Loss of Offsite Power (successful auto-start of the Emergency Diesels);
- Loss of Feedwater Heating;
- Turbine Trip with By-Pass;
- Load Reject with By-Pass;
- Main Transformer Fire; and Loss of Normal Heat Sink (Isolation Condenser Actuation).

Generally, plant equipment has adequately responded to these events. Where equipment malfunction exacerbated recovery from the event, assessments were promptly performed and corrective actions implemented, e.g., Feedwater Flow Control, or improved maintenance practices have been implemented, e.g., 4 kV breakers.

3.6 Special Verifications and Improvement Initiatives

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

- Assembling design and licensing information and improving its accessibility;
- Revising or establishing more specific calculations that implement the design bases (which facilitates verification on an ongoing basis);
- Verifying that plant configuration and performance is consistent with design; and

- Establishing monitoring programs to confirm conformance with specific aspects of design on an ongoing basis.

Significant examples of special verifications and improvement initiatives are discussed in Section 3.6.1 through 3.6.6, below.

3.6.1 Assembling Design and Licensing Information and Improving its Accessibility

The improved accessibility of design and licensing basis information, and many of the supporting calculations which implement the design bases, have further enhanced Dresden's ability to maintain plant configuration and performance on an ongoing basis. Specific examples of activities to improve accessibility are addressed in Sections 3.6.1.1 through 3.6.1.6, below:

3.6.1.1 Design Calculation Turnover, Indexing, and Control

As described in Appendix I to this response, ComEd implemented a program which transitioned the design control function from the NSSS supplier and Architect Engineer to ComEd and developed inhouse ComEd engineering capability. As part of this effort, ComEd acquired calculations used by the Architect Engineer in the design of many of the SSCs important to safety. For Dresden, approximately 12,000 calculations have been acquired. These are indexed, controlled in ComEd's Electronic Work Control System (EWCS), and available at the station. This effort was completed in November 1996.

The control and availability of calculations was noted as a significant weakness at Dresden by the 1996 NRC Independent Safety Inspection. Accordingly, several commitments were made during the inspection to obtain additional calculations, improve indexing, and improve control. These issues are discussed in greater detail in Action (e).

3.6.1.2 UFSAR Review Program

The UFSAR Review program was discussed in detail in Action (b), Section 2.6.1. The UFSAR and operational procedure review and corrective actions resulted in an improved UFSAR and conformance between plant operation and the UFSAR.

3.6.1.3 Upgraded Technical Specification Review

The Technical Specification Independent Review Group (TSIRG) was established to perform a comprehensive, line-by-line review of Dresden's proposed Upgraded Technical Specifications. The objective was to identify concerns that could result in non-compliance with the Upgraded Technical Specifications upon implementation. This effort is discussed in Action (b), Section 2.6.3

In addition to the procedure enhancements resulting from this effort, numerous Technical Specification comments were generated, including administrative items, technical questions, and setpoint changes. Several new Setpoint Calculations were performed to support the effort.

As a result of these reviews, greater assurance exists that the station is configured, maintained and operated in accordance with its design and licensing bases.

3.6.1.4 Vendor Technical Information Program (VETIP)

The availability of up-to-date vendor technical information, e.g., vendor technical manuals, bulletins, letters, etc., is essential to the proper installation, maintenance, operation, testing, calibration, troubleshooting and repair of SSCs. In 1990, in response to Generic Letter 90-03, ComEd implemented VETIP to control the update and distribution of vendor technical information. Lack of program ownership and inconsistent station support hampered effective implementation. As a result of failures in the station's 4 kV breakers in the spring of 1996, the station examined the existing VETIP program and embarked on a three year effort to ensure compliance with regulatory requirements. Under this program, vendors periodically (triannually) provide updates to their manuals, including notification if no changes have occurred since the last report. A key element of that program is clear ownership and accountability. In 1997, Dresden expanded the vendor recontact program to include certain non-safety related equipment in the VETIP effort.

Upon receipt at the station, the vendor information, i.e., technical manual change, is reviewed by the VETIP group and routed to appropriate system and component engineers for action. As part of the technical review, a specific check for "system interaction or design basis interface impacts" is required. Procedure review and plant walkdown also occurs. This review process is required to be completed within 90 days of receipt of the vendor information.

Proper execution of the VETIP program provides assurance throughout the life of the plant that components and systems are configured, maintained and operated in accordance with its design and licensing basis.

3.6.1.5 Equipment Data Base Improvements

A number of initiatives have been implemented at Dresden to improve the accuracy and availability of design basis information contained in various engineering databases. Some of the more significant improvement initiatives are discussed below in Sections 3.6.1.5.1 through 3.6.1.5.3.

3.6.1.5.1 Component Tracking and Control

An important element of configuration control is the knowledge of manufacturer and model number information for all components installed in the station. Previous control mechanisms consisted of Material Engineering databases, maintenance databases, Parts Evaluations, engineering databases (Instrument, MOV, Fuse List, etc.); however, recent projects combined

most of this data into the Electronic Work Control System (EWCS) database, thus centralizing and providing consistent control of the information. To date, over 60,000 components have manufacturer and model number data entered into the EWCS system. With the use of the EWCS system, design verified data can be identified, substitute replacements can be identified and component Bills-of-Material can be established. Revision of the approved manufacturer and model number data is now controlled by the design change process in the EWCS system.

Knowledge of proper manufacturer and model number data assists the Dresden Work Analysts in specifying replacement parts, provides tight integration between components and their respective controlled vendor manuals, assists in investigations of industry issues, and provides the opportunity for tight engineering control of plant configuration. Dresden's knowledge of the installed components, and thus our confidence in conformance with the design and licensing bases, has been greatly enhanced by this program.

3.6.1.5.2 Classification of Structures, Systems and Components (SSCs)

Proper safety classification of structures, systems and components (SSCs) is required to provide solid design bases as to the safety significance of the components. Dresden station currently controls the classification of structures, systems and components in the EWCS system for components and the Master Equipment List (MEL) for structures and systems. The following fields are included in the component classification scope: 1) Safety Class; 2) Seismic Category; 3) Electrical Class; 4) Environmental Qualification; 5) ASME Quality Group; 6) Regulatory Guide 1.97; and 7) Operating Mode.

Deficiencies in the component classification scope were identified in the mid 1980s and a classification upgrade program began in 1991 for Dresden. Under this program, over fifty systems and subsystems were reviewed and over 30,000 components classified. The program provides a classification for the seven main fields listed above and documents the basis for the classification on individual Component Classification Worksheets, which are maintained in Component Classification Binders. The binders include color coded drawings depicting safety system boundaries. The MEL was the primary control mechanism for component classification until September 1996 at which time the Electronic Work Control System (EWCS) was populated with the classification data and control was transferred to EWCS. The MEL continues to control the structural classification, position papers, system review status and other miscellaneous descriptive classification information which is not applicable to EWCS database control. The component classification project and controls established have greatly increased the completeness and reliability of the classification data which is used daily by plant personnel in performing work. A coordinator manages the program and a process exists for station personnel to resolve classification questions and issues. Additionally, classifications are kept current by requirements in the design change process. In the Station's view, the above classifications have led to greater understanding of the Dresden design and licensing bases.

3.6.1.5.3 Instrumentation Data Design Controls

Instrumentation design is controlled using the instrument data sheets, drawings, calculations and procedures. The Electronic Work Control System (EWCS) database currently controls instrument identification, classification, setpoints and various parameters and attributes, which are summarized in data sheets. Drawings control the contact arrangements and logic. Station procedures control instrument calibration.

The instrument data sheets were originally controlled on hard copy drawings. The instrument data sheet (IDS) project began in 1988 at Dresden to create the IDS electronic database to maintain design data formerly maintained on the data sheets. This project involved walkdown of over 10,000 instruments to verify nameplate data and location. It also included a review of critical plant drawings, station procedures, and Technical Specifications; and incorporation of existing hard copy data into the electronic design control system. The IDS database was formally implemented at Dresden in 1993 with over 21,000 records. In performing the various reviews, many inconsistencies were identified and corrected. The project was coordinated with the Operations Department in labeling over 7,000 instruments in the plant. In September 1996, the IDS data was transferred to the EWCS database which then became the database of record for instrument data.

Having this information in a single database allows quick resolution of design and licensing basis issues. Due to the data cleanup efforts which accompanied these migration efforts, greater confidence exists that the station's configuration matches the design and licensing bases, and that the station is being operated in accordance with those bases.

3.6.1.6 Control Room Drawing Improvements

Control of Critical Control Room Drawings (CCRDs) is maintained through a site administrative procedure. Dresden has significantly improved the process of maintaining and updating the CCRDs over the last three years. A CCRD coordinator from Design Engineering is assigned to update and maintain the most current status of the CCRDs. Revision of CCRDs is no longer being done by other Architect Engineers. Revision of CCRDs is exclusively processed at the station utilizing computer aided drafting techniques, thus reducing the turn-around time to as low as 24 hours. CCRDs are updated prior to the associated design change being released for "Operations Use." CCRDs affected by Temporary Alterations (Temporary Modifications) are processed within 24 hours of its installation or removal. A monthly surveillance is conducted to ensure that the CCRDs reflect the most current status of Temporary Alterations involving CCRDs.
System walkdowns were also performed in 1994 and 1995 to identify discrepancies between installed equipment and associated Process and Instrumentation Diagrams (P&IDs), including CCRDs. Over 70 CCRDs were revised to correct identified discrepancies.

As a direct result of the strengthened CCRD program (including walkdowns) there is reasonable assurance that Control Room Operators have access to accurate drawings for important plant systems.

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

Numerous programs have been conducted at Dresden to improve the quality of important design basis information. A few of the more significant programs are described in Sections 3.6.2.1 through 3.6.2.5, below.

3.6.2.1 Setpoint Control Program

The setpoint control program was developed to ensure consistency between design bases and instrument setpoints. ComEd developed a standard instrument database, along with a standard methodology for performing the supporting setpoint calculations. Additionally, controlling procedures were enhanced to ensure translation of required instrument settings into the field and subsequent gathering of performance data for future setpoint adjustments. Significant efforts were undertaken to improve the instrument database, validate and as-build the data contained, and create setpoint calculations that accounted for appropriate instrument setting inaccuracies and tolerances within the required design bases. A variety of programs are underway which will assure that critical setpoints are supported by a sound calculational basis. These efforts include: 1) A review of calculations performed under prior programs; 2) Technical Specification Upgrade Project (see Section 3.6.1.3); 3) Instrument Performance Monitoring (see Section 3.6.2.2); and 4) The design basis reconstitution project for the twelve most risk significant systems (see Action (e), Section 5.4).

3.6.2.2 Instrument Performance Monitoring

The station is currently developing a program to monitor and assess Instrument Drift and Out-of-Tolerance. This program will give the station the capability to evaluate the acceptability of Instrument Setpoints (as-found and as-left performance data). This program (like others noted in this Action) also provide the station with opportunities to identify significant deviations between information contained in the design and licensing bases and from equipment installed in the field.

3.6.2.3 Generic Letter (GL) 89-10, Motor Operated Valves (MOVs) Program

In order to provide adequate assurance that safety related motor operated valves would function in accordance with their design bases, the NRC issued GL 89-10 and supplements requesting industry evaluation and testing of MOVs. To meet the requirements of GL 89-10, ComEd documented the design bases for safety related MOVs, reconstituted calculations, established performance requirements, and performed comprehensive static and dynamic testing of MOVs against the performance requirements; ComEd has also adjusted MOV setpoints, modified equipment, and revised operating and maintenance practices as necessary to ensure that safety related MOVs will reliably perform their intended function under design bases conditions. Ongoing implementation of this program, including performance monitoring and trending, were established through procedural controls. A program coordinator was established in Engineering

to oversee and evaluate MOV test results and to ensure that ongoing actions are taken as needed to continually validate and assure acceptable valve performance consistent with design bases.

Although the results of the ComEd MOV program are nearly fully incorporated into other ComEd design basis programs, better interface arrangements between the MOV program and other station programs are being pursued to ensure that the MOV design bases are maintained in the future. These programs include the Electrical Load Monitoring System (ELMS) process for voltage drop calculations, the Load Monitoring System (LMS) process for pipe loading, and the modification process for system changes.

In December 1996, the NRC conducted the close-out inspection for GL 89-10 and found Dresden in compliance with all applicable regulatory requirements and commitments. As a result of the GL 89-10 program, the station has confidence that the safety related MOVs will perform all required design and licensing bases functions.

3.6.2.4 Electrical Load List Control and Voltage Setpoint Calculations

Prior to the late 1980s, Dresden Station used a manual load list to control 480 volt motor loads on the plant auxiliary power system. Although the manual list was adequate for the original plant design, additional loads have been added and system capacities are more closely approached. Two databases were developed to provide better control of electrical loads: Electrical Load Monitoring System (ELMS)-AC and ELMS-DC.

Electrical Load Monitoring System - AC

Station auxiliary power system loading is tracked by an engineering database created in the early 1990s to evaluate, track, and control electrical load on the AC distribution system. The database calculates short circuit current, bus voltage, and cable and distribution equipment loading on both a system wide, and individual bus basis. For any changes in auxiliary power system loading, these values are then compared to the established acceptance criteria for a given bus or load and those outside the established criteria are flagged, requiring more specific analysis. Control and tracking of the database is governed by ComEd procedures. During the initial runs of the output reports, the calculated loads were unusually high. As a result, field measurements were taken and used as an interim basis for determining auxiliary power system adequacy. A field monitoring program is being implemented to obtain actual operating data for use in establishing an ELMS-AC diversity factor. Applying this diversity factor will enable a more realistic modeling of the electrical auxiliary system.

During the Dresden Electrical Distribution System Functional Inspection (EDSFI) (see Section 3.7.1.3), the adequacy of the AC distribution system was closely examined. Several deficiencies were found in the ability of equipment to function under degraded voltage conditions. Significant effort was expended in assuring that those loads required under accident conditions would have adequate voltage under degraded voltage conditions. Modifications were performed to increase the size of several bus interconnecting feeds and replace control power transformers to

assure proper control circuit operation. A plan to upgrade the ELMS-AC program was also initiated following the EDSFI (expected completion is mid 1998), specifically including:

- Creation of a master database: Previously, when calculations were performed for various plant operating conditions (or special studies) a new database was created. As plant loading changed over the years, the various databases that existed to support the various calculations were not always updated to reflect the latest loading. With the use of the master database, all ELMS-AC calculations will be working from the same file which reflects the latest plant loading.

- Revision of the Diesel Generator loading calculations to incorporate the master database.

- Revision of the Electrical Auxiliary Distribution System calculation to incorporate the master database.

- Revision of the Degraded Voltage Analysis calculations to incorporate the master database. Improvement in the modeling of selected unit auxiliary system database components and verification of assumptions were included in this revision.

- Walkdown all 480 Volt switchgear and motor control centers against the key diagrams and ELMS-AC load tabulations for confirmation of services assigned to the bus, and to confirm the load characteristics.

- Modification of the ELMS-AC connection ampacity for 480V switchgear Motor Control Center (MCC) feeds to be the current feeder breaker setting. Any MCC load changes will be compared to this value and will be flagged if the rating is exceeded. Breaker settings have been identified and will be incorporated in the next scheduled revision of the ELMS calculations.

Electrical Load Monitoring System - DC

The ELMS-DC computer program is used to track loading on each safety-related stationary battery. This program, utilizing its databases, calculates cell size per IEEE-485, and performs battery voltage profile calculations. Control and tracking of the database is also governed by ComEd Nuclear Engineering Procedures. This effort provides added assurance that the electrical distribution system in the plant is in accordance with its design bases. An improvement effort similar in scope to the ELMS-AC effort is also underway and is expected to be completed by the end of 1997.

Significant effort has been expended at Dresden to provide assurance that electrical loading data accurately and conservatively reflect the Station's design and licensing bases. Significant improvement in the accuracy of the ELMS databases has been made since the NRC's EDSFI noted significant discrepancies. When the ELMS improvements are complete, an accurate modeling of Dresden electrical systems will exist, providing greater assurance that the plant is

operated in conformance with its design and licensing bases. Completion of the ELMS upgrade programs is tracked in the Nuclear Tracking System (NTS).

During the 1996 NRC Independent Safety Inspection, an error in battery loading assumptions was determined while reviewing the 250 V battery calculation which significantly reduced its reserve margin. A modification to incorporate a minor time delay in the cycling of a DC valve was made to resolve the issue. Additionally, several relatively minor errors were discovered in the 125 V battery loading calculation. In the aggregate, these loadings resulted in a slightly reduced margin in the 125 V battery capacity requirements. These issues are tracked in NTS and are being resolved.

3.6.2.5 Cable Tray Loading

The Dresden Station Cable Management System maintains a database for cables, cable raceways and related subjects for plant electrical installation design. These programs document, calculate and control various engineering and design issues related to electrical installation to assure the cable design is in compliance with the established criteria. These issues include: 1) Cable routing separation criteria; 2) Cable Tray fill; 3) Cable Tray static loading; 4) Power Cable thermal loading; 5) Safe Shutdown Cable routing capability; and 6) Cable combustible loading per Fire Zone.

During the Electrical Distribution System Functional Inspection (EDSFI) (see Section 3.7.1.3), a calculation error was identified in the portion of the cable management system used to determine power cable ampacity. As a result of this error, a significant number of cables were found to be potentially thermally overloaded. Upon discovery of the error, an initial test was performed on the worst case cable tray and it was determined that no cables within that cable tray were actually thermally overloaded. Further testing was then performed on a larger population of cable trays. This testing was done to determine a conservative means of quantifying geometric and system diversity to allow the cable management database to more accurately calculate allowable cable thermal ampacity. This action resolved approximately thirty percent of the apparent overloads.

A significant number of the apparent over loaded cables was caused by inaccurate data. By obtaining this data, the number of remaining overloaded cables was reduced to approximately 350 cables. Resolution of the remaining cables by more detailed case-by-case analysis is ongoing, is tracked in NTS and is expected to be complete in late 1997.

During the 1996 NRC Independent Safety Inspection, several minor issues arose regarding consistency of cable length in cable loading calculations. These issues are being addressed. When the cable tracking system improvements are complete, a more accurate modeling of Dresden electrical systems will exist, providing greater assurance that the plant is operated in conformance with its design and licensing bases.

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

Special verifications and resulting programmatic control and data improvement initiatives have been implemented over the years as a result of industry lessons learned. Some of the more significant initiatives and the results are presented in Sections 3.6.3.1 through 3.6.3.7, below.

3.6.3.1 Fuse Control Program and Fuse List

Audits identified weaknesses in the fuse control process at several nuclear stations, including ComEd. As a result, safety related fuse walkdowns were begun in 1993 to ensure that the installed fuses were consistent with the design. Where discrepancies were noted, the installed fuse was evaluated and determined to be acceptable or replaced. ComEd developed a standard fuse database (from which the fuse list was generated), along with a standard engineering process for fuse selection and control

The fuse control program at Dresden Station is a two phase activity, i.e., a walkdown and a maintenance phase. Dresden Station is currently undergoing the transition between first and second phases. In the first phase a known population of approximately 7,000 fuses was identified. Of these, Dresden has walked down approximately 5,600 fuses. The results of the walkdowns have been compiled into a Fuse database, which was recently transferred into the Dresden Electronic Work Control System (EWCS).

An independent review of these 3500 fuses showed that less than 10 % of these fuses have some conflict either between the drawing and the as-built configuration, or between the drawing and the fuse label in the plant. Most of the conflicts were between the drawing and the labeling. Significantly, none of the 5,600 fuses were found to be sufficiently out of specification to require replacement to assure the integrity of the electrical system which they protected. No significant problems were identified as a result of the walkdowns or the independent reviews performed.

Walkdowns for the remaining readily accessible fuses will be performed as equipment becomes available. As a result of this effort, there is a high level of confidence that Dresden's electrical system have adequate over-current protection. With the transfer of the fuse list database into EWCS, confidence exists that the configuration will be maintained in accordance with design and licensing bases.

3.6.3.2 NRC Inspection and Enforcement Bulletin (IEB) 79-02, "Pipe Support Base Plant Designs Using Concrete Expansion Anchor Bolts"

The scope of this effort included verification that baseplate flexibility had been adequately addressed in design calculations, verification of adequate factor of safety between anchor bolt design load and bolt ultimate capacity, definition of design requirements pertaining to cyclic and seismic loading conditions, and performance of a Quality Control documentation search verifying that design requirements had been met for cyclic loads and embedment depth.

ComEd performed inspections and static, dynamic, and relaxation load testing on all accessible safety related piping systems. The IEB 79-02 program concluded that the design approach pertaining to concrete expansion anchor plates at Dresden was adequate. The effort also identified that sufficient design margin existed in the design of expansion anchor assemblies to accommodate the effects of the flexibility of base plate assemblies.

Dresden completed its IEB 79-02 program and transmitted its results to the NRC in July 1981. Concurrence of compliance with IEB 79-02 requirements was obtained from the NRC in July 1986.

A ComEd Nuclear Station Work Procedure controls the installation of expansion anchors providing minimum embedment depth, spacing, angularity, and torque requirements for the expansion anchors for which the design is based. Safety related expansion anchor installations are Quality Control verified and documented on installation checklists.

3.6.3.3 NRC Inspection and Enforcement Bulletin (IEB) 79-14, "Seismic Analyses for As-Built Safety-Related Piping Systems"

In 1979, IEB 79-14 was issued to all power reactor facility licensees and construction permit holders, to verify that the seismic analysis applied to the actual as-built configuration of safety-related piping systems. At Dresden, the safety related piping includes Seismic Category I systems as defined by Regulatory Guide 1.29 and UFSAR. The action items that follow apply to all safety-related piping 2.5 inch diameter and greater and to Seismic Category I piping, regardless of size, which was dynamically analyzed by computer (or an alternate analytical method) in cases where walkdown information indicated a deviation from the original design. It was required that the actual configuration of these safety-related systems meet design requirements. The basic requirement of the IEB 79-14 program was to demonstrate that the piping systems meet FSAR commitments, to assure that the system would perform its intended function in the event of a design basis earthquake.

Detailed re-analysis of the piping and supports for all FSAR load combinations were performed when walkdown information indicated a deviation from the original design. Dresden piping was re-analyzed in accordance with USAS B31.1, 1967 and requirements specified in the FSAR. Exceptions are: 1) the Air-Containment Atmosphere Dilution (ACAD) and Containment Atmosphere Monitor (CAM) small bore piping which was designed and installed to ASME Section III criteria; and 2) the Dresden Unit 3 recirculation system that was designed and fabricated to ASME Section III, Class 1 criteria but installed to USAS B31.1 criteria. Supports were evaluated for loads imposed by the piping system. Supports were qualified, modified, or newly designed to ensure that the piping and supports complied with the FSAR and applicable Codes. During the field walkdowns, the following items were obtained: 1) Piping geometry (pipe length and size, valve location, bend radius); 2) Support locations and orientations; and 3) Insulation type etc.

In response to IEB 79-14 program, ComEd has performed evaluations of safety related piping systems per IEB 79-14 requirements. As a result, new supports were added when required, some

of the existing supports were modified, and numerous other supports were qualified for significantly higher loads than the original design loads.

ComEd has evaluated safety related piping operability concerns whenever a discrepancy is discovered between design documents and the actual plant configuration. This includes a representation of different types of piping configurations and support discrepancies. The majority of these discrepancies are due to damaged supports, inaccurate walkdown input, and pipe modeling errors. The quantity of errors has not been significant when compared to the scope of the IEB 79-14 program. Additionally, numerous audits have been performed on the program by ComEd and the NRC that have demonstrated that Dresden's program is acceptable and consistent with industry practice.

As a result of the IEB 79-14 program, ComEd is confident that safety related piping meets applicable seismic design criteria and that there is a high degree of fidelity between piping drawings and installed piping.

3.6.3.4 Boiling Water Reactor (BWR) Mark I Containment Upgrade Program

In 1975, the NRC transmitted letters to all utilities owning BWR facilities with the Mark I containment system design, requesting that the owners quantify the hydrodynamic loads and assess the affect of these loads on the containment structure. The letters reflected NRC concerns about the dynamic loads for Safety Relief Valve (SRV) discharges and the need to evaluate the containment response to the newly identified dynamic loads associated with a postulated design basis Loss of Coolant Accident (LOCA). The re-analysis of the Mark I piping and subsequent modifications were completed in the early 1980s. A large number of Mark I supports were modified. In 1987, a piping verification program was performed to verify the integrity of the post Mark I piping program. This program was similar to the original IEB 79-14 program. The scope included as-built walkdowns. As a result, some piping had to be reanalyzed and/or reconciled to resolve discrepancies. Several new supports were added and existing supports were reinforced to meet requirements. The piping now meets code requirements. Additionally, the numerous audits performed on the program by the ComEd, NRC and other third party organizations have demonstrated that Dresden's program is acceptable and consistent with industry practice. An Inservice Inspection (ISI) program (see Section 3.4.2.1) is in-place at Dresden which inspects safety related supports per the approved ISI program.

Dresden's Mark I containment program helps assure that the station is configured, maintained, and operated in accordance with applicable design and licensing basis criteria.

3.6.3.5 NRC Inspection and Enforcement Bulletin (IEB) 80-11, "Masonry Walls"

In response to IEB 80-11, ComEd reviewed the adequacy of Class I masonry walls, i.e., those walls supporting Class I equipment or components, and specific non-Class I walls whose structural failure may affect Class I equipment or components. Class I items are those required to withstand the effects of the worst case postulated earthquake and remain functional. Extensive field walkdowns were performed to determine the as-built configuration of the walls and

attachments from equipment or components to these walls. Modifications, where necessary, were made.

In December 1982, the administrative procedure for initiating and processing a work request was revised to include a masonry block wall inspection checklist. An NRC inspection was conducted in April 1983 with no items on noncompliance or deviations being identified. In May 1983, the NRC found that the report submitted for Dresden to be complete, thorough and adequately address the concerns of IEB 80-11. Dresden continued in implementing controls of attachments (or equipment adjacent) to masonry walls. Signs were posted on the safety related masonry walls in October 1989. In addition to the posted signs, control is maintained via administrative procedures. The masonry block wall inspection checklist has been updated several times since 1982. This checklist must be completed during the preparation of a work package affecting either safety related block walls or safety related equipment near these walls. If any of these items is checked yes, then engineering must be contacted. In addition, there is a "Masonry Block Wall Inspection" item in the Design Change Implementation Checklist. The Master Equipment List was updated to include all safety related masonry walls in February 1993. Follow-up actions on masonry wall #38 were completed in August 1992 when the safety related cabling for the Reactor Water Cleanup System was relocated from the masonry wall.

As a result of a review in preparation for this 10 CFR 50.54(f) response, discrepancies were discovered in the identification of safety related walls. Corrective actions will be taken utilizing the Integrated Reporting Program.

3.6.3.6 NRC Generic Letter (GL) 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46"

In general, the active mechanical and electrical equipment of older plants, was not designed to the more sophisticated seismic testing and analysis standards applied to newer installed equipment or to the newer licensed stations. The NRC issued GL 87-02 requesting verification that equipment would withstand a postulated seismic event and function as required. In response to GL 87-02, a seismic review was initiated in accordance with the Seismic Qualification Utility Group's (SQUG's) Generic Implementation Procedure (GIP). The GIP approach relies on developing a Safe Shutdown Equipment List (SSEL) which identifies equipment needed to achieve and maintain safe hot shutdown conditions as defined by a plant's Technical Specifications. This equipment is then seismically reviewed in accordance with the GIP methodology. An assessment can be made concerning the seismic adequacy of each item on the SSEL by observation and evaluation during plant walkdowns. By evaluating seismic demand criteria, selected caveats to ensure similarity to the GIP equipment classes, an anchorage evaluation, and a seismic interaction proximity assessment, the trained walkdown engineer can be satisfied that the equipment will survive the plant's design bases seismic event.

At Dresden, a total of 428 equipment items, including tanks and heat exchangers, were seismically evaluated. Of this population, 112 items were declared equipment outliers. Some equipment had more than one outlier issue associated with it; therefore, the actual number of outlier issues to be

resolved is 164. Of these 164 issues, 63 have been resolved. Approximately 5600 relay contacts were evaluated. Of these, 1058 contacts in 533 individual relays were declared to be outliers. All cable tray and conduit raceways in areas containing safe shutdown equipment were walked down and checked against the inclusion rules and other seismic performance concerns as specified in the GIP. A total of ten outliers resulted from the cable and conduit raceway reviews. Of the ten outliers, five have been resolved analytically. All remaining outliers have been assigned Nuclear Tracking System (NTS) numbers and will be completed within two refueling outages per unit following receipt of the NRC's Safety Evaluation Report on the USI A-46 submittal.
The SQUG effort provides confidence that plant equipment required for safe shutdown will meet its seismic design bases operability criteria.

3.6.3.7 Safety Related Contact Testing

The Safety Related Contact Testing Adequacy (SRCTA) project of 1990 to 1992 was initiated in response to an "unlanded electrical lead" incident and other inadequate surveillance testing incidents at several ComEd stations from 1985 to 1989. Seventeen Dresden Station systems were reviewed:

Low Pressure Injection/Containment Cooling
Standby Gas Treatment/Reactor Building HVAC
High Pressure Coolant Injection (HPCI)
Automatic Depressurization (ADS)
Primary Containment Isolation
Neutron Monitoring
Reactor Protection (RPS)
Control Rod Drive (hydraulics only)
Standby Liquid Control
Core Spray
Emergency Diesel Generators (EDG)
Isocondenser
Anticipated Transient Without Scram
Residual Heat Removal/Shutdown Cooling
Refueling Bridge Interlocks
Torus to Reactor Building Vacuum Breakers
Condenser Pit Level

A detailed review of all applicable schematic diagrams and surveillance procedures for each system was performed to determine the adequacy of contact testing, ensuring positive verification in the open and closed state of each contact. A system summary report was prepared for each system that included a listing of all functions, schematics, procedures, untested contacts and other testing concerns identified during the review. Contacts identified as untested were resolved with special testing and procedure changes as necessary. In addition, all function, reference, schematic, procedure, contact function, and contact test status information for each contact of each system was entered into the SRCTA databases. These databases are currently being updated in Dresden's response to GL 96-01, "Testing of Safety-related Logic Circuits."

As a direct result of this program, Dresden is confident that safety related logic circuitry is designed, configured and tested in accordance with the station's design and licensing bases.

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

In addition to routine monitoring activities previously discussed, special programs have been implemented and diagnostic tools used at Dresden to provide enhanced monitoring of key system components. This provides added assurance that performance remains consistent with design bases. Two significant examples of such special programs include the Emergency Diesel Generator (EDG) reliability testing and the GL 89-13 programs described below.

3.6.4.1 Emergency Diesel Generator (EDG) Reliability Program

In 1993, the EDG reliability program was implemented at Dresden under Nuclear Operations Directive NO-TS.20. The EDG reliability program requirements are based upon the Station Blackout (SBO) rule; Regulatory Guide 1.155; Regulatory Guide 1.9, Revision 3 and NUMARC 87-00. The program is required to maintain and monitor EDG reliability over time for assurance that the selected target EDG reliability levels are being achieved. The EDG reliability program includes the following: 1) monitoring EDG reliability against target reliability levels (trigger values); 2) requirements for comprehensive condition monitoring, surveillance testing, maintenance, root cause analysis, problem close-out and information services; and 3) actions required if EDG reliability falls below the target levels (or exceeds trigger values).

The following elements are also an integral part of the EDG reliability program.

- A ComEd Engine Analysis program has been developed to improve EDG predictive and preventative maintenance. An EDG reliability monitoring program for properly monitoring and tracking EDG reliability against SBO target levels (95.0%) has been developed.

- A Diesel Engine condition MONitoring (DEMON) program has been developed to properly monitor EDG operational characteristics and trends.

One of the main focuses of the EDG reliability program is to closely monitor the EDG system including every EDG demand to ensure that the Station's EDG licensing basis is maintained. Dresden Station has seen continued excellent reliability performance and improved consistency.

3.6.4.2 NRC Generic Letter (GL) 89-13, Service Water

GL 89-13 required Dresden 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 a program was implemented to monitor heat exchanger effectiveness and overall system performance on an ongoing basis. Implementation of this program provides added assurance that essential service water systems can function in accordance with their design bases.

The design of service water systems was reviewed in accordance with GL 89-13, Action Item IV. The review was completed in September 1992. The Containment Cooling Service Water

(CCSW), and Diesel Generator Cooling Water (DGCW) systems were identified as meeting the Generic Letter's definition of service water systems. The review was accomplished both through a review of design documents and procedures and through a walkdown. Aspects of the review included system configuration, flood protection, pipe supports, emergency power supply, and functional logic evaluation. Special consideration was given to the single failure analysis during the configuration review.

One major deviation from the design bases was found in the DGCW system configuration. It consisted of isolating the system from supplying water to emergency air coolers for the ECCS pump cubicles. During the SWOPI this was found to be a design change in progress. Justification for the change was already established, and actions to implement the change were identified. These changes have been implemented and this configuration is now part of the station's design and licensing bases. Several minor discrepancies were also discovered and corrected between the design documents and as-built configurations. It was concluded that the service water systems will perform their intended functions in accordance with the licensing basis for the plant. Only one action from GL 89-13 remains to be completed. This consists of a performance test on the LPCI/CCSW Heat Exchanger. Plans are underway to accomplish this performance test.

The adequacy of Dresden's conformance to GL 89-13 was evaluated by the NRC in the 1993 Service Water Operational Inspection (SWOPI) (see Section 3.7.1.5). The inspection results were generally favorable; however, it found that some programs resulting from GL 89-13 were not being implemented in a timely manner. Except for the LPCI/CCSW Heat Exchanger test, these issues have been addressed.

3.6.4.3 Performance Centered Maintenance (PCM) Program

The objective of the PCM program is to improve the overall material condition of the plant by selecting effective and appropriate preventive maintenance activities for the station's components and equipment. The PCM concept focuses on predictive maintenance techniques with the goal of performing maintenance on equipment when needed.

The PCM process starts with identification and prioritization of plant systems and component groups using the Maintenance Rule (10 CFR 50.56). These systems are then examined in detail to ensure all components are identified and appropriate PM tasks are being performed. PM tasks are modified, added or deleted to meet the goal of improving the effectiveness of the PM program.

As a result of the PCM process, work instructions are modified or produced to support field performance of new or revised PM tasks. During preparation of the work instructions, plant drawings, system and component walkdowns, and vendor information reviews are performed. Occasionally, these reviews uncover discrepancies in information between plant components, plant drawings, equipment labels, component nameplate data and the engineering database. The discrepancies are documented and resolved. The reviews also ensure that identified material is proper for each PM task.

The PCM program, which is still in development, has increased the accuracy of the engineering database and provided greater assurance that the station is configured and maintained in accordance with its design and licensing basis. Additionally, the PCM program will help assure that plant equipment is being properly maintained and will perform in accordance with its design and licensing bases.

3.6.4.4 Generic Letter (GL) 89-08, Flow Accelerated Corrosion (FAC) Inspection Program

To meet the requirements of GL 89-08, Dresden is implementing a comprehensive program to inspect all highly FAC susceptible single phase and two phase high energy carbon steel piping and thus preclude failures in this piping. Large bore piping is modeled in the EPRI "CHECWORKS" program and small bore piping is handled administratively in the "Susceptible Non-Modeled Program." This effort contributes to plant safety and improves design basis conformance by ensuring that piping systems function as designed and do not corrode to an unacceptable pipe wall thickness.

Due to the FAC inspection results and the better understanding of FAC degrading mechanisms, several plant modifications have changed the piping component design, routing and material selection to avoid pipe wall thinning, before it could lead to catastrophic failure or unscheduled repair or replacement.

Inspections at Dresden indicated that several turbine extraction steam and feedwater heater nozzles have experienced FAC, and repairs have been performed where required. Some small bore piping systems have experienced FAC. Replacements have been made, where required. As a result of experience with FAC in small bore piping, the small bore inspection program will be expanded during the next refueling outage (D3R14).

3.6.5 UFSAR Rebaseline

Dresden initiated a UFSAR rebaseline project in 1991. The rebaseline project examined significant correspondence between the NRC and ComEd. Special attention was paid to issues associated with the Systematic Evaluation Program. Design changes were also reviewed to assure correct incorporation into the UFSAR. The program also reformatted the UFSAR to meet current regulatory guidance. The project was completed in 1993 and the Rebaselined UFSAR became effective in January 1994. Two limitations of the UFSAR Rebaseline effort were: 1) it did not include a review of plant procedures and field walkdowns; and 2) it did not attempt to "validate or reconstitute" basic design inputs, e.g., calculations. The review of the UFSAR against plant operating procedures was subsequently performed in 1996 and is discussed in Action (b), Section 2.6.1. Although some errors were discovered and corrected from the initial rebaseline effort, e.g., inconsistency in primary containment locked valve lists, the outcome of the rebaseline effort is that there is very high confidence that the UFSAR accurately represents the licensing bases of the station. The 1996 NRC Independent Safety Inspection Team noted that the rebaseline effort was a significant upgrade.

3.6.6 Design Basis Document (DBD) Development and Validation

ComEd realized in the late 1980s that the station needed to have better understanding, access, and control of design information. A DBD management group and a DBD task team were established to develop a DBD program. Dresden developed system and topical DBD priority lists to determine which DBDs were most needed. The main DBD writers were the original designers, from the NSSS vendor and AE.

A critical aspect of assembling DBDs was how to determine what design information was the critical "design basis" information. The NUMARC 90-12 Design Bases Program Guidelines were used to do this and became the basis of program procedures that were issued to define the process of preparation, review, issuance, revision, and control of DBDs and open item evaluation. The resulting purpose of ComEd DBDs was established:

> The DBD shall identify the design input requirements, their rationale and bases, and design analysis for selected structures, systems, and components or topics. The DBD does not establish new design input requirements or design analysis but is intended as an assembly of information that already exists. The DBD presents an "overview" of key design input requirements and design analysis and also serves as a "source reference" by pointing the user to the applicable design document(s) for additional detailed information.

The scope of DBD preparation did not include any plant walkdowns, detailed validations, or reconstitution of design inputs or analysis, since plant configuration and calculations were considered to be acceptable. However, the scope did include identifying and resolving any conflicts in design information between any existing design documents. Reviews by Plant Engineering, Operations, and Training Departments would serve to provide reasonable assurance that operating and testing procedures agreed with design bases requirements.

The DBD development process was controlled by a Writer's Guide, which provided guidance to the writers for consistent format and content. The process included identifying original plant design bases, incorporating changes resulting from various types of modifications, reviewing existing design information, and resolving any conflicts between documents. The NSSS and AE writers accessed ComEd databases in addition to original design bases documentation and subsequent modification files to arrive at the current design bases. The current design bases were reviewed to identify and resolve potential discrepancies with the original design. Technical Specifications and the FSAR were reviewed during the preparation of the DBDs to identify and resolve potential discrepancies between design and licensing basis documents.

Reviews were performed by ComEd organizations and other AEs that were involved in the design and operation of the station. The ComEd groups included Site Engineering; System Engineering; Corporate Engineering; Nuclear Fuel Services; Mechanical and Structural Design; Electrical, Instrumentation and Control Design; and the Site Training departments. This provided a check to ensure the latest design information was identified.

To date, twenty-three DBDs have been generated for Dresden, including:

* # High Pressure Coolant Injection
* Automatic Depressurization System
Shutdown Cooling
* Auxillary Power (4 kV and 480 V AC)
Seismic Design and Analysis (Topical)
Control Room HVAC
Nuclear Boiler Instrumentation
Single Failure Criteria (Topical)
Reactor Protection System
Standby Liquid Control System
* # Low Pressure Coolant Injection
Neutron Monitoring System
Reactor Recirculation System
* 125 and 250 Volt DC
Emergency Diesel Generator
Core Spray
Primary Containment
* Feedwater, Condensate and Reactor Level Control
Design Basis Events (Topical)
Primary Containment Support Systems
Standby Gas Treatment System
* Isolation Condenser
* Containment Cooling Service Water

* = Probabilistic Risk Assessment (PRA) identified "Risk Significant System"
# = Initiation logic of these systems is identified by the PRA to be Risk Significant

Plans are being developed to create another ten DBDs. These additional DBDs will assure that the twelve most Probabilistic Risk Assessment (PRA) "Risk Significant Systems" will have DBDs. The additional DBDs on Risk Significant systems include: 1) Turbine Building Closed Cooling Water; 2) Main Steam Safety and Relief Valves; and 3) Service Water. To date, one DBD has been validated. The other twenty-two existing DBDs, as well as the new DBDs, will be validated. The DBD program has resulted in enhanced information regarding Dresden design and licensing bases, and provides greater assurance that important SSCs are in substantial compliance with those bases.

In addition, a pilot validation of the Standby Gas Treatment (SBGT) System DBD was performed in May 1996. The purpose of the validation was to assess the overall effectiveness of the DBD development program, assess the quality of the DBD and validate that the design basis requirements (as documented in the DBD) were consistent with the physical plant and plant operating documents and practices. The six areas investigated as part of this effort were: design bases implementation; design control and system modification; operating procedures; maintenance procedures and testing; training materials; and issues management. A validation checklist was prepared and contained focused activities in each of the these six areas. The validation effort was supported by system walkdowns. The validation requirements in the checklist were developed from specific requirements contained in the DBD, as well as various plant documents.

Ten open items were identified and additional enhancements and minor inaccuracies in the DBD were identified and are being resolved. The results indicated that the overall quality of the DBD was acceptable. The design bases of the system was properly implemented in the plant. The open items were screened at the time of validation for significance. No operability or safety issues were identified. These items are being evaluated and resolved by appropriate cognizant engineers.

3.6.7 Latent Material Condition and Material Condition Improvement Initiative

In 1995 a team of third party industry experts performed an extensive review of key Dresden systems to determine, based on industry experience, material conditions issues which had not yet manifested themselves in open plant performance. The identified issues were compiled and prioritized. On going system improvement plans were developed in 1996 to address the more significant issues. These prioritized issues are evaluated in the station's business planning process.

3.7 Audits, Inspections, and Configuration/Performance Assessments

As described in Sections 3.2 through 3.6, there is a substantial basis on which to conclude that Dresden SSC configuration and performance is, in the vast majority of cases, consistent with the design bases. Processes are in place to control design changes, to monitor plant configuration and conformance, and to restore nonconforming or degraded equipment. Special initiatives, such as development of DBDs for twenty-three systems, a UFSAR rebaseline in 1991, and a UFSAR review against plant procedures in 1996, provide high confidence that design basis information is addressed.

This section discusses special "one-time" assessments at Dresden, which are defined as in-depth reviews of programs or some aspect of design, construction, maintenance of SSCs which are conducted by either ComEd, the NRC or a third party. Over the years, these assessments have identified program weaknesses, including a recurring weakness in the area of timeliness in resolving issues. ComEd has attempted to address these weaknesses as they have been identified. These assessments and corrective actions provide further bases on which to conclude that the SSCs are currently properly configured. More importantly, ComEd initiated some significant steps in late 1996 to verify current assurance that the plant can be safely operated and to generally improve the use of design documentation. This section first discusses some of the historic assessments and the actions taken. Following this is a discussion of several initiatives to verify the current plant condition and to improve the quality, maintenance, and accessibility of design information.

3.7.1 Audit and Inspection Findings

3.7.1.1 The Dresden Safety System Outage Modification Inspection (SSOMI)

From December 1985 through July 1986, the NRC conducted a Safety System Outage Modification (SSOMI) of the Dresden Unit 3 Recirculation Pipe Replacement Project. A follow-up inspection was conducted in 1988. In addition to specific deficiencies, the NRC noted that it would be "prudent to perform a review of previous modifications -- even on a sampling basis -- in order to provide additional assurance that plant safety margins, from a system functionality stand point, have not been inadvertently degraded during the modification process."

In response to this stated concern, ComEd developed a four element program. First, ComEd conducted a thorough review of the NRC SSOMI concerns. This review addressed the safety significance of each concern as well as the overall safety significance of the concerns when

reviewed collectively. Second, ComEd performed a supplemental inspection of additional modifications not reviewed by the NRC to identify any unacceptable plant-wide trends. Next, a Safety System Functional Inspection program (SSFI) (discussed in Section 3.7.1.2) to apply vertical slice audit techniques to systems other than those reviewed by the NRC was developed. Finally, a Safety System Functional Testing Program was developed.

Due to uncertainties with the reliability of the Diesel Generators at Dresden identified by the ComEd self-initiated SSFI, and questions over the degradation of critical safety systems due to lack of control over modifications, an Architect Engineer was engaged to design a comprehensive testing program to prove the functional design adequacy of the Diesel Generators and their support systems. An extensive test program was performed, the results were analyzed, and the generic implications of identified discrepancies were documented. Changes were made in design control processes to correct weaknesses. The Diesel Generators were determined to meet the specifications of the original design, of the FSAR, and of the SER.

3.7.1.2 Early Safety System Functional Inspections (SSFIs)

In 1987, the Quality Assurance (QA) Department began to perform SSFIs based on the NRC's Inspection Plan (IP) 98401 at ComEd's four older stations. In all, eleven systems were examined. A QA department procedure was developed that included the elements of the NRC's IP. Auditors were trained, outside technical experts were contracted to assist in the inspections, and a schedule was established. Systems were selected mainly for their safety significance and the opportunity to observe multi-discipline, modified systems. By choosing this sample, the underlying processes that controlled the design, procurement, installation, test, operation, modifications, repair and replacement of components could be assessed. The typical inspection team was composed of six to ten engineers -- at least one of whom was an SRO -- supported by two to four inspectors, and was conducted typically over a four to twelve week period. Heavy emphasis was placed on consistency and agreement of design documents, drawing, and procedures through all phases of modification and maintenance, and on verification of physical conformance of the plant to the design documents. All deficiencies were tracked to closure and appropriately verified as complete and closed. In accordance with the corporate procedure, as apparent generic deficiencies were identified, they were included in future inspections at all sites for further validation of the system wide process weakness.

From May 1987 to August 1987, Dresden's Emergence Diesel Generators were examined. Twelve deficiencies and ten concerns were identified and resolved. From May 1988 to August 1988, Dresden's High Pressure Coolant Injection (HPCI) and 125 VDC systems were examined. Four deficiencies and thirty-six concerns were identified and resolved. The number and significance of these deficiencies and concerns were consistent with the deficiencies and concerns identified at the other three ComEd stations.

In addition to the specific issues presented above, an analysis by corporate Nuclear Engineering, an SSFI-experienced consultant, and QA resulted in a number of generic or programmatic issues. These generic issues were resolved, along with the specific SSFI issues.

At Dresden, questionable operability and post maintenance testing of the Unit 2/3 Diesel Generator and potential for common mode failure for Diesel Generator circulating water supply, was identified. All systems were eventually evaluated as being operable and capable of performing their intended functions, although document changes were necessary for disposition.

3.7.1.3 Electrical Distribution System Functional Inspection (EDSFI)

An NRC inspection team conducted an EDSFI in the summer of 1991. The objectives of the inspection were to determine if the electrical distribution system was capable of performing its intended safety function as required by its design and licensing bases. The EDSFI identified weaknesses in the areas of degraded voltage relay setpoints, design calculations, documentation of cable ampacity, DC system design, design documentation for mechanical systems supporting the emergency Diesel Generators, root cause analysis, post modification testing, and fuse control. Each of these items has been the focus of improvement initiatives, as well as continued scrutiny by the NRC in the years following the EDSFI, including the recently completed 1996 NRC Independent Safety Inspection. Specific improvement initiatives have been previously described related to the Electrical Load Monitoring System (Section 3.6.2.4), Cable Tray Loading (Section 3.6.2.5) and the Fuse Control Program and Fuse List (Section 3.6.3.1).

Although some of these areas continue to require attention, significant overall improvement has been realized since the EDSFI. Increased fidelity between the plant configuration has been experienced, and the significance of deviations identified (as measured by operability assessments) has decreased.

3.7.1.4 Vulnerability Assessment Team (VAT) Evaluation

A Vulnerability Assessment of Dresden Station was conducted by a team of experienced industry personnel during the period from April 20, 1992 to July 10, 1992. The Nuclear Operations Division Executives established an Assessment Planning Team (APT), as part of ComEd's "mature plant improvement initiative." The purpose of the APT was to arrange for and provide guidelines for the performance of an objective, qualitative risk assessment of selected systems and their vital components. The systems selected for review were those judged to be of most significance in reducing core melt frequency. Certain generic issues and on-going engineering programs were also selected for review because of their impact or potential impact on the selected systems.

To conduct the assessment effort, a team of six technical personnel was assembled. The basic objective of the assessment was to identify vulnerabilities affecting the selected systems and determine if timely corrective action was being taken. The assessment process included a review of database information and other documentation; interviews of Maintenance, Operations, Technical Staff, Engineering, Construction, and consultant personnel; and non-intrusive walkdowns. Approximately 3000 work history entries and work requests were screened, 100 open Nuclear Tracking System (NTS) items were reviewed, and 200 system interviews were conducted.

A total of seventy-six vulnerabilities were identified by the VAT. Most, though not all, of the vulnerabilities noted had been previously identified; however, a number of instances were noted where timely corrective action had not been initiated or completion dates had been allowed to slip. The VAT noted, "While it is difficult to draw general conclusions without knowing all of the factors, including resource limitations, that went into establishing or revising a priority, the VAT found that increased attention to follow-through, was warranted." Based on the guidance provided for the assessment, the VAT believed that sufficient systems had been reviewed to qualitatively assess the station condition against potential issues or deficiencies which could increase the probability of core damage. The VAT found that the review of further systems would not, in its view, be likely to identify emergent issues which would be of greater significance or have greater need for additional management attention. Rather, the VAT believed that timely follow-through to address each of the vulnerabilities identified should be given a high priority.

It was believed that 63 of the original 76 identified vulnerabilities have been resolved; however, the 1996 NRC Independent Safety Inspection Team identified that two of the VAT vulnerabilities had been closed without proper resolution. The remaining thirteen have schedules for resolution and are being tracked in the NTS. The VAT was another example of an extensive examination of plant systems. Although the focus of the VAT examination was on issues that could lead to a core damaging event, the resolution of issues of vulnerability such as those identified by the VAT contributes to confidence that important structures, system and components will be able to perform their design and licensing basis functions.

3.7.1.5 Service Water Operational Performance Inspection (SWOPI)

An NRC Service Water System Operational Performance Inspection (SWOPI) was conducted in April 1993. The inspection included a limited mechanical design review; detailed system walkdowns; review of system operation, maintenance and surveillance; and corrective actions related to GL 89-13 (see Section 3.6.4.2). The overall conclusion of the inspection team was that Dresden's Service Water System design and operation were effective; however, several weaknesses were also identified including: slow implementation of GL 89-13 guidelines; deficiencies in the IST Program; and narrowly focused corrective actions.

The results of GL 89-13 efforts and the SWOPI provide confidence that the Dresden Service Water System is configured and operated within its design and licensing bases. The more general performance concerns, particularly regarding timeliness, are being addressed as discussed in Section 3.7.3.

3.7.1.6 1996 NRC Engineering and Technical Support (E&TS) Inspection

The 1996 NRC E&TS inspection was conducted by a team of six NRC and NRC contractor personnel over a four week period. Positive comments were made on the Dresden design change process. Some issues were identified including: problems with post modification testing, minor errors in calculations, and isolated examples of inadequate 10 CFR 50.59 evaluations. In general, it was a positive inspection. No violations of NRC requirements were discovered.

3.7.2 Verifications and Improvements

ComEd recognizes the performance and configuration issues that have been raised over the years for Dresden Station, and particularly in 1996 by the NRC Independent Safety Inspection team. Both immediately preceding and following these assessments, ComEd has initiated actions to address weaknesses and verify current conditions. These actions provide further basis for the conclusion that there is presently reasonable assurance that the Dresden Station is substantially in compliance with the design bases, and that the plant can be safely operated.

3.7.2.1 Engineering Design Control Self-Assessment

In July 1996, a team of senior ComEd engineers and experienced industry experts was assembled to perform a self-assessment of the engineering design control processes. The purpose of this self-assessment is to:

- Confirm the technical adequacy of a safety system with high PRA impact and confirm the consistency among the design bases, Technical Specifications, procedures, design documentation, and the physical plant (using engineering criteria from NRC Safety System Functional Inspection Procedure (IP) 93801).

- Assess the effectiveness of the design control process and the adequacy of the procedures and management controls that implement the process.

- Demonstrate ComEd's ability to perform effective self-assessments (in accordance with the general guidance of IP 40501 - Licensee Self-assessment Related to Team Inspections).

- Germinate effective techniques of engineering self-assessment.

The Engineering Design Control Self-Assessments were completed in September 1996 on the Low Pressure Coolant Injection (LPCI) and Containment Cooling Service Water (CCSW) systems. The team concluded that the LPCI and CCSW systems are capable of performing their safety functions as established by their design bases, licensing requirements and commitments. Further, the team concluded that engineering performance is adequate in the conduct of routine and reactive activities and providing technical support to other Dresden departments.
The team expressed one significant area of concern: Site management had a tendency to lack follow-up to significant problem solutions and issues and commitment closure, e.g., number of required CCSW pumps and availability of adequate LPCI pump Net Positive Suction Head. This backlog issue is discussed in Section 3.7.3.

3.7.2.2 Assessment of Current Conditions and Engineering Controls

In November 1996, in response to the NRC Independent Safety Inspection, ComEd completed and documented to the NRC the results of a series of reviews to verify that current plant

conditions are safe and support continued operation, and that engineering controls are adequate. These reviews included:

- A review of twelve safety significant systems. This included an evaluation of current system surveillance and acceptance criteria with respect to design functions as identified in the UFSAR. In each case, it was determined that surveillance results establish that the equipment will operate as expected and will perform the intended safety function.

- A review of three significant plant transients between 1989 and 1996. This concluded that safety systems called upon to operate performed as required.

- An ongoing screening of key parameters on twelve systems most important from a risk perspective. This screening includes a review of key operating parameters against existing system calculations. The critical parameters at each system have been identified. It is scheduled to be complete by February 1997.

In addition, ComEd committed to several additional actions, including actions to verify the UFSAR and expand and validate the DBDs. These actions are discussed in Action (e). In addition, ComEd established the Engineering Assurance Group described in Action (a).

The actions taken to date, coupled with the processes and prior assessments previously described, provide assurance that the plant is currently configured and operated in a safe manner.

3.7.3 Backlog Reduction

In August 1995, in response to a large number of "Configuration Management" PIFs, a panel of station personnel was commissioned to examine configuration management issues at Dresden. The team examined the following areas: 1) Setpoint Control; 2) Vendor Equipment Technical Information Program (VETIP); 3) Instrument Data Cards; 4) UFSAR; 5) Fuse List; 6) Temporary Alterations; 7) Drawings; 8) Open Modification Backlog; 9) Master Equipment List; 10) Electronic Work Control System; 11) Field Change Requests; 12) Design Calculations; 13) Design Bases Documents; 14) Operations Caution Card Log; 15) Out-of-Service Program; and 16) Plant Labeling.

The team found significant problems. They recommended detailed root cause investigations in five of the areas reviewed and effectiveness reviews on corrective actions recently implemented in five other areas. These root cause investigations were conducted, corrective actions assigned, tracked and implemented. One of the resulting investigations was a Level 2 Investigation examining: 1) Organizational Interfaces; 2) Process Interfaces; and 3) Training and Qualification. Significant program revisions were created as a result of the investigations, e.g., the Alternate Parts Replacement program for replacing obsolete equipment.

These investigations provided the impetus for the Dresden configuration management backlog reduction program. This program is addressing approximately fourteen separate configuration

management backlog reduction efforts. One of the more important is the Open Modification Backlog Reduction Program which has reduced the number of old backlog modifications from 239 to 21 over the last fifteen months. This effort was viewed as a strength by the 1996 NRC Independent Safety Inspection Team.

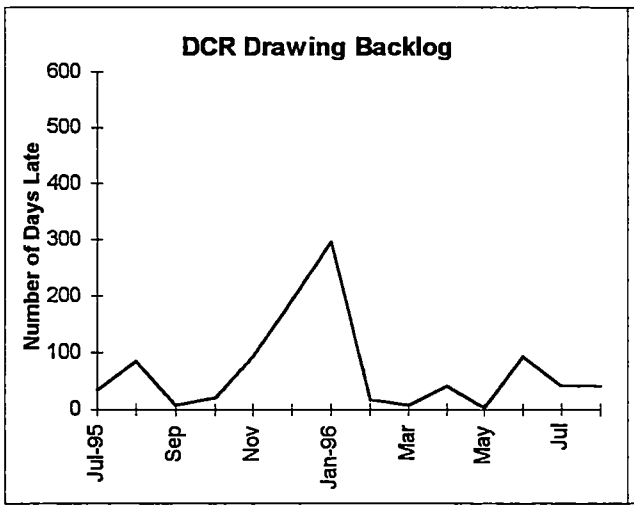
Large backlogs of engineering work and timeliness of completion have been recognized problems at Dresden. Plans have been developed to reduce or eliminate the backlogged items. Extensive performance monitoring programs have been implemented since 1995. The following items are monitored:

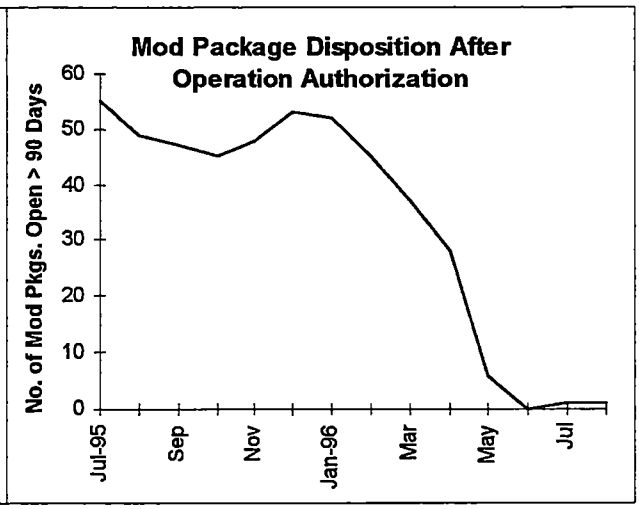
- Drawing Change Requests (DCRs) Open beyond their target closure (usually 90 days)
- "Operations Authorized" Design Change Packages (DCPs) open beyond their target closure (90 days)
- Open Non-Outage Related Temporary Alterations (the goal is less than 20 open)
- Vendor Equipment Technical Information Program (VETIP) manuals in technical review greater that their target (90 days)
- Open Operator Workarounds (OWAs)
- Open Engineering Requests (ERs)
- "Old Modification" Backlog Reduction
- Instrument Data Sheet Backlog Reduction
- Component Classification (Master Equipment List (MEL)) Backlog Reduction
- Original Design Basis Document (DBD) Project Completion
- Target Drawing Improvement Programs
- Fuse List Program
- Setpoint Calculation Improvement Program
- Environmental Qualification Binder Update Backlog Reduction Effort
- Electronic Work Control System (EWCS) Improvement Efforts
- Calculation Turnover (Architect Engineer to ComEd) Project
- Control Room Work Requests
- Various Quality Indicators
- Number of Open Nuclear Tracking System (NTS) Items (which includes NRC commitments, among other areas)
- Number of "Overdue" and "Extended Due Date" NTS Items
- EQ Binder Updates

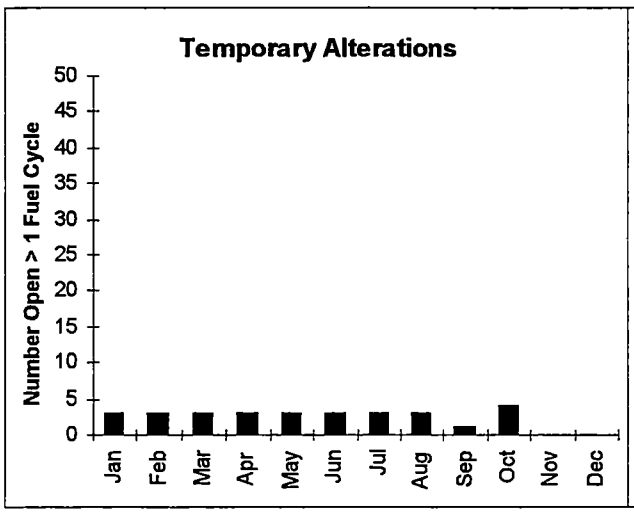
There have been significant reductions in the targeted areas. A sample (the first six items above) of the monitoring charts are included in this section. The backlog reduction efforts and associated monitoring programs are important elements in Dresden's efforts to address timeliness of implementation of corrective actions.

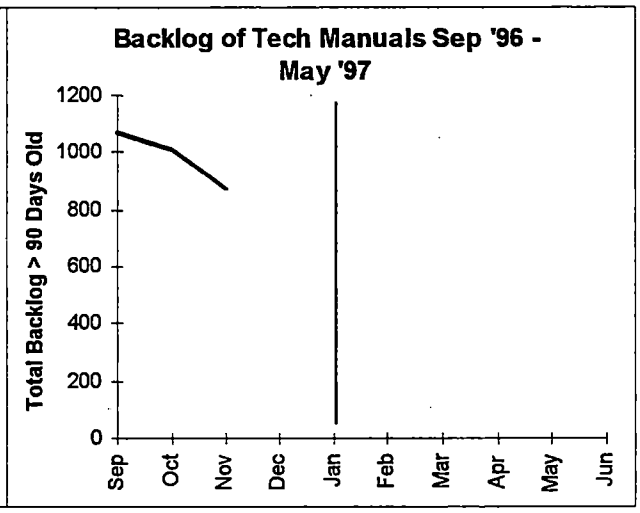
Furthermore, the historic weakness with respect to timeliness relates largely to issues with perceived low safety significance. With the more recent management focus on reducing backlogs, the Engineering Department reviews backlogs monthly to confirm that there are no high priority issues that remain unresolved. This approach provides reasonable assurance that historic

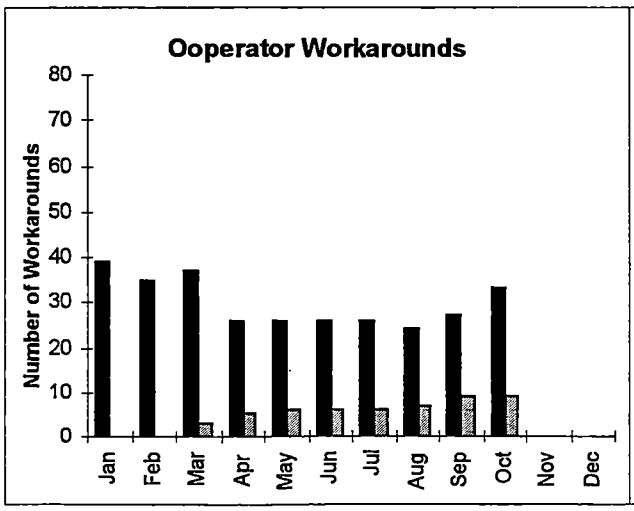
timeliness weaknesses have not resulted in current discrepancies between the plant and the design bases. The end result will be continuous improvement of sound programs governing station design and licensing bases, with a low number of backlog configuration management issues.

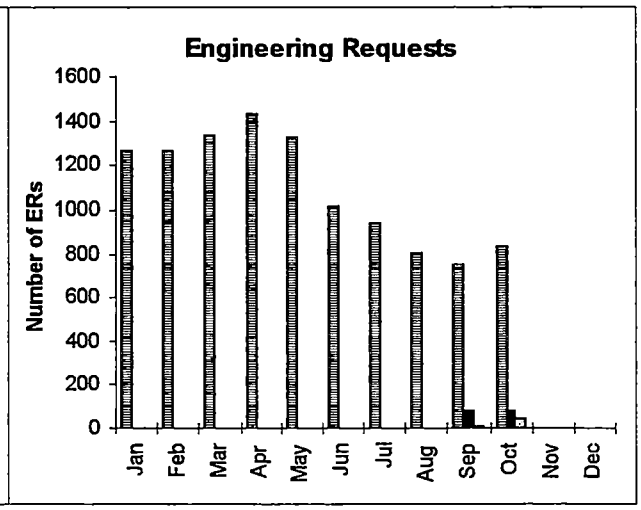












3.8 Conclusion

Notwithstanding historic performance weaknesses, ComEd concludes that there is reasonable assurance that the Dresden Station structures, systems and components are configured and operated in substantial compliance with the design bases. This conclusion is expected to be verified by the scope of work outlined in the January 30, 1997 T. J. Maiman letter to A. B. Beach.

Since initial licensing, the plant operation and configuration have been controlled by procedures that require consideration of the design bases. In addition, the configuration and operability of equipment is continuously confirmed by ongoing surveillances, verifications and operating experience.

Furthermore, numerous targeted special verifications and improvement initiatives have been conducted over the years. These efforts have resulted in improvements, where necessary, and in the aggregate provide assurance as to the current consistency between design documents and the plant. In particular, there have been initiatives to assemble design information and improve its accessibility, to collect and to verify design basis information (including a UFSAR rebaseline review and development of DBDs), to revise or establish more specific design basis information, and to confirm equipment conformance with the design bases. When equipment configuration or operation problems are identified -- which is to be expected from time to time -- a process exists to control restoration.

As a result of the 1996 NRC Independent Safety Inspection, ComEd undertook additional reviews for twelve risk-significant systems. These reviews confirmed that current plant conditions are safe and support continued operation.

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 Dresden Station to identify problems, to determine the root cause and the extent of the problem identified, and to report problems to the NRC. It also describes the processes used to resolve identified problems through implementation of appropriate corrective actions, including actions to prevent recurrence. In addition to the corrective action program, this section also addresses targeted reviews, audits and inspections that have led to the identification of problems outside of the normal corrective action program.

In summary, processes and procedures in place at Dresden Station require site personnel to identify conditions adverse to quality, including non-conformances in design documents, as-built plant drawings, and procedures. The Station uses an Integrated Reporting Program (IRP) to meet the requirements of 10 CFR Part 50, Appendix B. As discussed in more detail below, the IRP provides an integrated method for identifying problems, investigating such problems through a controlled process, and controlling and tracking corrective actions. Audits and inspections are performed periodically to verify compliance.

Station processes ensure that design concerns are formally documented and evaluated. Identified conditions adverse to quality are assessed to determine their impact on operability in accordance with NRC Generic Letter 91-18. A process is in place to identify and resolve conditions adverse to quality. Root cause investigations are performed for significant conditions adverse to quality, and actions to prevent recurrence are tracked to completion. Station processes are also in place to identify and correct any generic implications. Controls are also in place for tracking, reviewing, and evaluating temporary alterations and operability evaluations for degraded conditions. Station processes require that identified conditions adverse to quality be evaluated for reportability to the NRC and, if appropriate, reported to the NRC pursuant to NRC regulations.

4.2 Integrated Reporting Program (IRP)

The IRP process is used by Dresden Station to meet the requirements of 10 CFR Part 50, Appendix B. The most important component of the IRP is the use of the Performance Improvement Form (PIF) for identifying problems. The PIF process is simple to use and is available for use by all personnel on site to report any discrepancy, deficiency, or concern, real or perceived. The Dresden senior management aggressively encourages all site personnel to document any concerns. The existing IRP process is described in site procedures with the necessary forms which are simple for anyone to use. The IRP process is similar at the six stations; however, to facilitate uniformity of the six stations, a new ComEd Nuclear Station Work Procedure (NSWP) is being developed and should be issued in 1997.

The PIF process includes provisions for the prompt identification and evaluation of operability and reportability concerns. Once a PIF is initiated, it is reviewed by the Shift Manager or

designee (an RO level) to determine whether an immediate nuclear safety or operability concern exists. If so, the Shift Manager takes appropriate actions to place the plant in a safe condition. All new PIFs are reviewed each shift. If additional input from engineering is required to demonstrate operability, then the issue is forwarded to Engineering for an "Operability Evaluation." Site procedures specify time constraints for completing operability evaluation.

All PIFs are reviewed by the Event Screening Committee (ESC), which normally meets on Monday, Wednesday, and Friday. Reviews are done in a timely manner to ensure that a safety concern is dealt with promptly.

The ESC evaluates the adequacy of any immediate corrective actions taken prior to it's review, determines if follow-up actions are required, determines if more information is required, and classifies the PIF based on safety significance. There are four levels of PIFs, from 1 to 4, with 1 being the most significant. All PIFs assigned a significance level of 1, 2, and 3 require root cause analysis. The ESC then assigns an action to the appropriate station organization for a Root Cause Investigation for levels 1, 2, and 3 PIFs.

PIFs are also entered into a database for trending and/or for informational purposes. The Station's PIF Coordinator is responsible for tracking PIFs and assigning actions in the Nuclear Tracking System (NTS) when corrective actions or root cause investigations are required. Trend analysts then review this data searching for adverse trends and common cause.

4.3 Other Processes That Identify Problems

4.3.1 Action Request (AR)

The Action Request (AR) process provides a mechanism for anyone on site to request an action. (It is generally described above in response to Action (a)). The AR is typically used for reporting routine maintenance issues on plant systems, structures, or components. The process is simple; the originator initiates an AR and hangs an Action Request tag on the equipment (if applicable). If the activity could potentially be an operability concern, procedures require that the Shift Manager be notified and that a PIF be generated. Action Requests are initiated and tracked via the Electronic Work Control System (EWCS).

4.3.2 Engineering Request (ER)

An Engineering Request (ER) is used as a method of requesting Engineering assistance and evaluation for a potential or existing problem. ERs are not intended to provide approval of work activities. Examples of activities for which an engineering evaluation is needed before corrective action may proceed include:

1. A problem in the plant requires a design change.
2. A problem in the plant requires a Temporary Alteration.
3. A valve installed in the plant does not match prints, and the correct valve must be ascertained.
4. An operational problem exists in the plant and requires a technical evaluation.

5. Exact replacement component, e.g., valves, gauge, motor cannot be obtained.

Engineering Requests are initiated and tracked via the Electronic Work Control System (EWCS).

4.3.3 Document Change Request

Discrepancies between plant documentation and the as-built condition of the plant can be identified through the DCR process which is described in the response to Action (a) and in Appendix II. A Document Change Request (DCR) serves two purposes. A "Turnover DCR" serves to update drawings subsequent to implementation of a design modification. An "As-Built DCR" serves to update drawings subsequent to identification of a discrepancy between the field and station controlled documents. An "As-Built" DCR is the mechanism for making a document change administratively based on an existing condition; no field work is performed. As-Built DCRs are reviewed via 10 CFR 50.59 screening to ensure that no unreviewed safety question exists.

4.3.4 Operating Experience Feedback (OPEX)

The OPEX program is the primary means used to review and evaluate operating experience information for applicability and determine necessary follow-up actions. This program applies to any source of industry operating experience information, including INPO SOERs and SERs, NRC Information Notices, Bulletins, and Generic Letters, Vendor/NSSS Bulletins, etc. In addition, significant PIFs and Nuclear Operating Notices from the other ComEd stations are reviewed by the OPEX program for applicability to the station.

Operating experience assessment and dissemination is the responsibility of the Regulatory Assurance Supervisor. OPEX information is processed through the site OPEX Coordinator. The OPEX Coordinator uses station procedures to determine whether technical review requirements associated with code, regulation issues, required, then a System Engineer (SE) is assigned. SEs review the information for actions required, prepare action plans as required, support the material through Onsite Review (OnSR) if necessary, assign actions as appropriate, track the items, process NTS items as necessary, and perform effectiveness reviews of selected items.

Regulatory information involving the NRC is processed by the Regulatory Assurance group. A review of the OPEX program was performed by the NRC during the recent Independent Safety Inspection (ISI), it was noted that although the backlog of Information Notices (IN) was reduced significantly from 90 to 18 open items, however, opportunities for improvement have been identified to make the processing and dissemination of information more rigorous. The station is currently in the process of identifying the required corrective actions to be implemented.

4.3.5 Operator Work-Arounds

The existence of an Operator Work Around (OWA) can also indicate the existence of a problem. An OWA is defined as a material or document deficiency which requires an operator to take compensatory or non-standard action to comply with procedures, design requirements, or Technical Specifications. An OWA is considered detrimental to good operating practice and

receives a high priority by Station management. Work Requests identified as OWAs receive special consideration during the scheduling process.

An OWA is identified as a problem and is processed through either the normal PIF process or the AR process. In either case, once it is identified as an operator work around, it is sent to the OWA coordinator. The coordinator screens the potential OWA, assigns a tracking number and prioritizes the item. The item is then assigned to an owner, usually the System Engineer. The System Engineer develops and action plans for resolution of the issue and processes ARs or ERs if required for design changes.

4.3.6 Technical Alerts

The Technical Alert program is a special ComEd program which identifies "lessons learned at one station" from operating experience feedback and making such insights available to the other stations. The Technical Alert content is designed to provide sufficiently detailed information on emerging engineering issues. In addition to lessons learned, it includes solutions identified and actions needed to address the issue at other locations. Since the inception of the program, seven Technical Alerts were issued in 1994, 15 in 1995, and 35 in 1996. Dresden evaluates the Technical Alerts for their applicability at Dresden and takes appropriate actions.

4.3.7 Nuclear Operations Notifications

A Nuclear Operations Notification (NON) notifies other ComEd Nuclear sites of a problem or event that has occurred at Dresden 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 an electronic mail account has immediate access to them. At Dresden, the Event Screening Committee (ESC) selects NON subjects from the PIFs they evaluate daily. If they determine that a NON should be published, it will be prepared by the cognizant station personnel and posted to the electronic bulletin board. With respect to "incoming" NONs, 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 Dresden Station.

4.3.8 Audits and Evaluations

Problems are also identified through the audit process, necessitating corrective actions where required. Examples of audit processes include:

4.3.8.1 Site Quality Verification (SQV) Audits

Since 1990, the SQV organization has conducted performance-based audits. This process change was implemented to enhance and strengthen the compliance-based audit approach used prior to 1990. Audits are conducted in accordance with Nuclear Oversight procedures and SQV instructions. The procedures and instructions establish the methodology, and the 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 more detail below.

4.3.8.2 ISEG

Independent Safety Engineering Groups (ISEG) are required only at post-Three Mile Island operating plants. Although not required, Dresden has formed an ISEG group. The group function is to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports (LERs), and other operating experience information, including information from plants of similar design, which may indicate areas for improving unit safety. As with SQV audits, the ISEG performs reviews in accordance with Nuclear Oversight procedures and SQV instructions. ISEG personnel also conduct surveillances of unit activities to provide independent verification that activities are performed correctly and that 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 more detail below.

4.3.8.3 Field Monitoring Program

Preparation for and conduct of field monitoring activities are also an SQV function designed 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 schedule is flexible and can be changed as necessary. Field monitoring activities typically include daily tours of the control room and witnessing field implementation of operating, test, or maintenance procedures or sequences. Deficiencies noted during field monitoring are documented on a Field Monitoring Report and a Corrective Action Record.

4.3.8.4 PIF Trending

The Trending Analysis in the SQV organization provides oversight by performing an independent analysis of station performance information in accordance with established guidance. Trends are reported to site 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. PIFs are initiated when adverse trends are identified.

Activities involved in performing trend analysis in support of the Corrective Action Program at Dresden Station include the following:

1. The data collected by the Integrated Reporting Process is classified and coded to allow searches for recurring problems and other conditions adverse to quality. This classification system ensures attention to significant issues and facilitates the identification of emerging issues. Codes are used to categorize PIFs into topical areas for analysis. For example, PIF data reported in Actions (a) and (c) of this response were pulled using this system.

    - Level 4 PIFs are coded with Proximate Cause Codes.
    - PIFs resulting in a root cause investigation (level 1,2,3) are coded with the actual cause codes as determined by completed root cause investigations.

2. Collection of Trend Analysis Data

Following are examples of typical areas from which the SQV analysts collect information for analysis:

- IRP database information.
- Information that is provided by the SQV audit, ISEG, and Quality Control groups.

3. Analysis of Data

The SQV analysts normalize trend data in order to provide a more realistic analysis, organize the material graphically, analyze the data, and identify the current trend. An adverse trend is defined as the occurrence of two or more of the following conditions. In determining whether an adverse trend exists, the following criteria are considered:

- Has there been an increase in the number of related events over an extended period of time?
- What is the significance and consequences of events, e.g., Regulatory, Budgetary, Nuclear Safety, Industrial Safety, Human Performance, Plant Operations, etc.?
- Are there deviations from industry standards or corporate/station goals?
- Are there events with common threads, e.g., common root causes, personnel, etc.?
- Have non-consequential or low consequential events occurred that are regarded as precursors of more significant events.

4. Reporting of Trend Analysis Results

The SQV Analysts periodically issue reports, i.e., monthly and quarterly, which identifies current trends in performance as directed by the SQV Director. The report distributed to appropriate station personnel, as well as to the NRC Senior Resident Inspector.

4.3.8.5 Quality Control Program

The Quality Control (QC) program is conducted in accordance with controlled site procedures. Non-conforming items, such as components, parts, spares, consumables, portable test equipment, and inspection and test procedures identified in the field, are documented via the PIF process.

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 team to create an atmosphere where employees can freely voice concerns. The individual raising the concern may request confidentiality and every effort will be made to assure the confidential status is maintained. Feedback will be provided to the individual raising the concern. If the individual does not agree with the resolution, the issue may be escalated to a higher level.

In addition, Dresden has created a Site Vice President (SVP) Hotline through which employees can freely voice their concerns directly to the SVP.

4.4 Other Processes That Determine Extent of Problems

There are several methods used at Dresden Station to determine the extent of identified problems and the corrective actions needed to address those problems. These methods include the following:

4.4.1 Root Cause Analysis

As part of the IRP program, 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 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 normally meets on Monday, Wednesday, and Friday, to review PIFs initiated since the last meeting. A root cause investigation or a proximate cause is assigned to each PIF depending on the severity of the issue as part of the screening/review process. Root Cause investigations, evaluations, and reports are conducted in accordance with site procedures. Root cause investigation techniques, checklists, and report format are provided in site procedures. Training is conducted for personnel performing root cause analysis.

4.4.2 Safety Evaluations

The Safety Evaluation process is described in general in Appendix II. The site specific safety evaluation procedure requires engineering personnel to evaluate the extent of condition of each change during the characterization of the change. Examples of items reviewed during the characterization of any change are as follows: the potential of the change to reduce or compromise the redundancy of existing systems; the effect of the operation of any other system on the system being evaluated, either directly or indirectly; power circuit interactions, electrical division separation, and the effects of the change on ventilation systems.

4.4.3 Operability Determinations

The operability determination process is described in general in Appendix II. Station operability determination procedure require that engineering personnel identify whether the problem, failure, defect, degraded or nonconforming condition impacts any required function of any affected safety related or significant non-safety related structure, system, or component.

4.4.4 OPEX Reviews

As discussed earlier in this section, the operating experience program procedures require that the response to the OPEX item address the impact on the station.

4.5 Other Processes That Identify and Implement Corrective Action

As discussed earlier (Section 4.2), the primary mechanism for initiating corrective action at Dresden Station is through the IRP process. For level 1, 2, and 3 PIFs, a root cause analysis is performed and appropriate corrective action is implemented. For level 4 PIFs, a proximate cause is assigned and occasionally a corrective action is developed, usually to correct the deficiency rather than addressing a programmatic issue.

Other problem identification processes identified above also have the potential to lead to corrective action. Examples would include station responses to SQV open CARs, OPEX information, responses to information from regulatory agencies and other third parties. All corrective actions are entered into the Nuclear Tracking System for tracking purposes.

4.5.1 Corrective Action Records (CARS)

The Corrective Actions Records 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. The CAR is used for documenting, reporting, follow-up, condition close-out and trending. There are four status levels and three levels of significance in the CAR program.

4.5.2 Nuclear Tracking System (NTS)

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

4.6 Other Processes Which Determine Lessons Learned

4.6.1 Root Cause Determinations

Station procedures require that root cause analyses be performed for significant conditions. The purpose of the root cause analysis is to identify the fundamental cause(s) of the condition. When

the fundamental cause(s) of the condition have been identified, corrective actions can be developed and implemented to prevent recurrence.

4.6.2 Effectiveness Reviews

Corrective actions implemented to prevent recurrence as a result of root cause analyses are reviewed for effectiveness in accordance with station procedures. The effectiveness review is typically conducted within three to six months of the implementation of the corrective action. In 1996, 109 effectiveness reviews were conducted. Approximately 80% of these have proven to be effective. Where not effective, additional actions are taken to resolve outstanding issues and another follow-up effectiveness review is performed.

4.7 Processes for Reporting Problems to the NRC

As problems are identified and screened as reportable to the NRC using the PIF process, actual notification requirements are delineated in site procedures which describe the process for reporting Station events as required by corporate directives, the NRC and other agencies, including the Illinois Department of Nuclear Safety (IDNS). Specific interpretive guidance is provided to the Shift Engineers, and others, in the ComEd Reportability Manual.

The reporting of material defects and non-compliances to the NRC per 10 CFR Part 21 requirements is controlled by site procedures.

4.7.1 Licensee Event Reports (LERs)

Station processes require that identified conditions adverse to quality be evaluated for reportability to the NRC and, if appropriate, reported pursuant to NRC. For example, issues that become PIFs are required by procedure to be reviewed for reportability. In addition to NRC regulations, guidance on reportability is provided in the ComEd Reportability Manual. This controlled manual provides an event driven system of decision trees to aid in reportability determinations. The Reportability Manual addresses notifications and reporting in accordance with 10 CFR 50.72, 50.73, 50.9 and Part 21 as well as other regulations. The Summary Tables contained in the Manual provide a concise encapsulation of the various reportability requirements.

4.7.2 Technical Issues Review Process

The purpose of the Technical Issues Review Process is to review technical issues, particularly those having generic implications, with respect to 10 CFR Part 21, "Reporting of Defects and Noncompliances." The process is implemented through a Technical Issues Review Committee or representative, who coordinates the actions to investigate and resolve technical issues, and provides guidance and/or solutions to engineering and licensing issues, particularly those common to more than one site.

Weekly meetings of the Technical Issues Review Committee are held, with all six stations participating via teleconference. Technical issues are identified for review from station events, vendor notifications, design concerns, Nuclear Network entries and other industry and regulatory

sources. The participation by each of the six stations and the corporate office provides multiple discipline reviews used in this process.

If a reportable issue is identified, the station management and VP-Engineering are involved in the review and if required, operability assessments involve independent reviewers.

4.8 Process Effectiveness

Some of the specific process elements described above are relatively new, such as 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 Dresden Station's history. Audits and assessments of these processes have been conducted by ComEd personnel and by external agencies, including the NRC. In general, these reviews have not identified any significant or programmatic deficiencies in the process procedures, their implementation, or the products. However, opportunities for improvement have been identified, and several improvement initiatives have been undertaken. For example, as described in Appendix I to this response, ComEd has created a six station peer group to develop a more common, improved Corrective Action Process. Additionally, 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.

4.8.1 Site Quality Verification (SQV)

During the last year, the SQV organization at Dresden Station has evolved into a more aggressive organization in terms of it's ability to identify issues to improve overall station performance. SQV's involvement in the area of Configuration Control has included a wide range of issues during 1995 and 1996. In fact, SQV was heavily involved in the initiation of five (5) of the seven (7) root cause investigation reports identified in the level in the level 2 investigation conducted by Station in the area of Configuration Control during August, 1995 (see Section 3.7.3). SQV's involvement in the five (5) of seven (7) root cause investigation associated with the level 2 investigation involving Configuration Control was as follows:

- In March 1995, SQV initiated a level 3 investigation into the area of Configuration Control in response to numerous PIFs generated on the issue, the investigation was initiated by SQV and others while serving as a member of the Event Screening Committee (ESC).

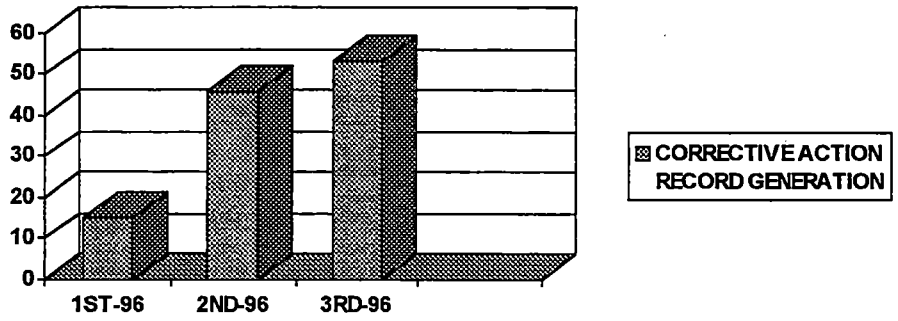
- The initiation of the level 3 investigation involving the issue of Temporary Alterations referenced in the level 2 investigation was led by SQV.

- SQV played a heavy role in the initiation of the level 2 investigation involving the issue of OOS program weaknesses. SQV's Trend Analysis Reports prior to the initiation of this level 2 investigation clearly indicated a problem in this area.

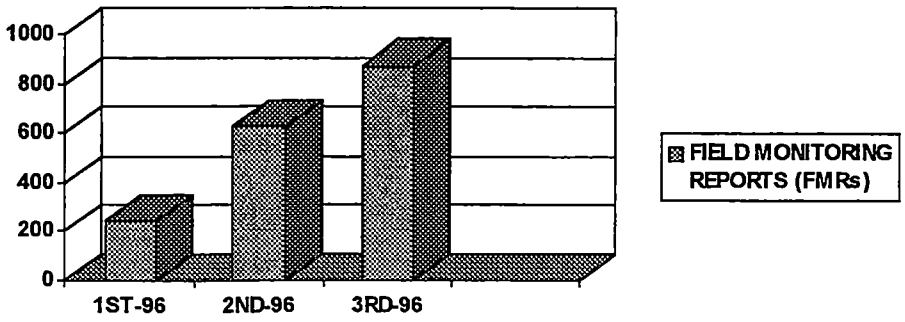
- SQV identified a deficiency during a surveillance of the Setpoint Change program in July/August 1995. A level 3 investigation on Setpoint Control and Reconciliation initiated in September, 1995.

- In September 1995, a level 3 root cause investigation was initiated by SQV on the effectiveness of the VETIP Program.

One factor contributing to this enhanced effectiveness is the increase in the diversity of the SQV individuals' backgrounds and expertise in areas such as Operations, Radiation Protection, Engineering, and Maintenance. Additionally, Senior Management and Supervisors with SQV experience were added to the SQV staff. Another contributor to this increase in effectiveness has been raising expectations within Dresden Station. This enhanced effectiveness is demonstrated by the following graphs and information:

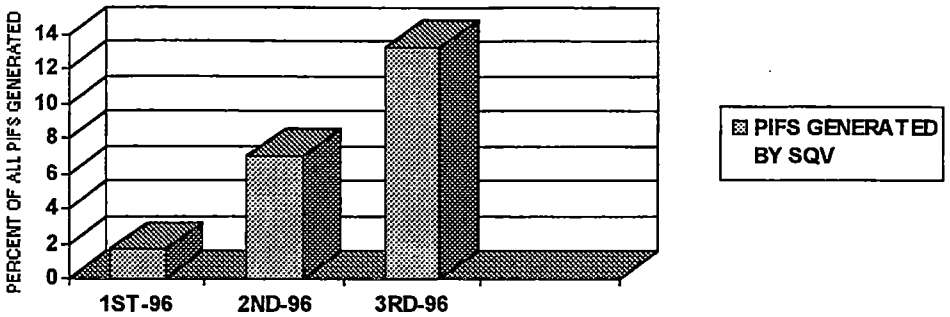


The Corrective Action Record generation rate has increased over 200 percent since the first quarter of 1996. Many of these CARs were generated during Field Monitoring which are real time field observation results. There has been an increase in FMRs in 1996. The increase in FMRs is noted in the graph below.



FMRs increased over 200 percent from the first quarter of 1996 to the third quarter of 1996. This increase in FMRs is also responsible for the increase in PIFs generated by SQV. The number

of SQV PIFs generated for 1996 as a percent of all PIFs generated for 1996 is represented in the next graph. The SQV monthly report provides a focus input for FMR coverage on adverse or declining areas, based on a review of the PIF data.



The percentage of PIFs generated by SQV in relation to all station PIFs generated has increased over 600 percent from the first quarter of 1996 to the third quarter of 1996.

In April of 1996, SQV developed a Station Self Assessment Procedure that identified which departments were required to perform self-assessments. Since development of this procedure, over 30 self-assessments have been performed. The self-assessments are evolving to critical line self-assessments. To date, 179 issues have been identified by the self-assessments and are being tracked in NTS.

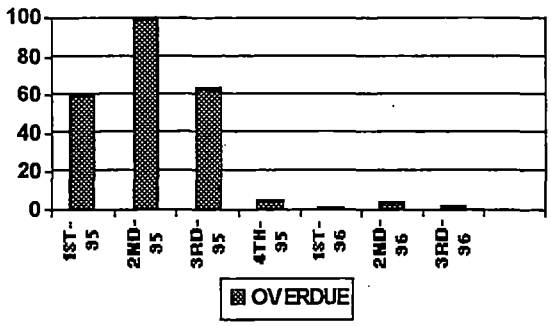
4.8.2 Corrective Actions Successes

Since the beginning of 1995, the Corrective Actions process has gone through numerous changes and corresponding improvements. Specific improvements include:

- Creation of a Corrective Action Group to administrate corrective action tracking, root cause analysis, PIF trending, and site self-assessment coordination.
- Creation of the Corrective Action Review Board (CARB) to review and approve root cause investigation reports.
- Independent review and approval of corrective actions prior to closure.
- Implementation of trained and dedicated root cause analysts.
- Increased implementation of corrective action effectiveness reviews.

Decrease in Overdue Corrective Actions

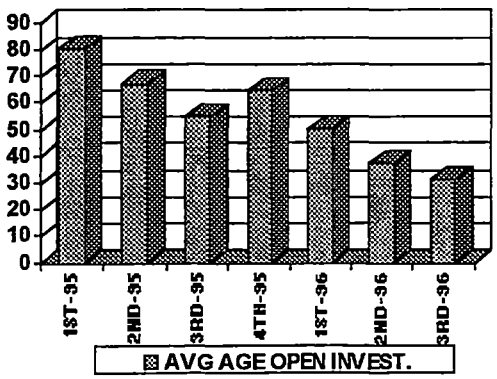
The number of overdue corrective actions has decreased dramatically since the beginning of 1995, as seen below. The number of items closed has steadily increased per quarter from 821 (closed the first quarter of 1995) to 1042 (closed the third quarter of 1996).



This is primarily due to an increased emphasis on accountability, mandated by station management, for overdue corrective actions. Since the beginning of 1996, there have been only nine overdue corrective actions. During this time period, 2853 NTS items were closed.

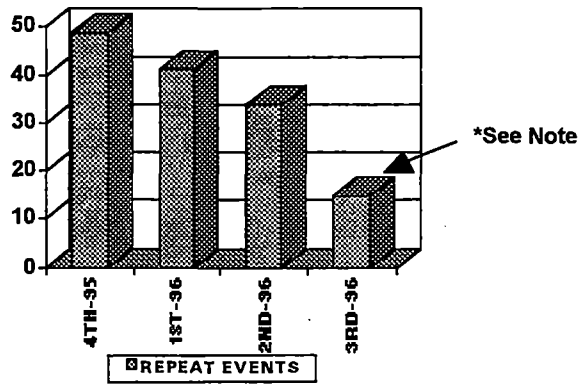
Decrease Average Age of Open Investigations

Since the beginning of 1995, the average age of open investigations has dropped considerably, from 80 days (in the first quarter of 1995) to 31 days (in the third quarter of 1996). This decrease in investigation time is due to a raising of station standards, and accountability with regards to the amount of time allotted for the performance of investigations, as seen in the graph below.



Drop in Repeat Events

Through more aggressive implementation of corrective actions and improved root cause analysis, the number of repeat events at Dresden have decreased as seen in the graph below. From the fourth quarter of 1995 to the third quarter of 1996, there was a 48% drop in the number of repeat events.



*Note: Repeat events have only been tracked since the last quarter of 1995. September 1996 repeat events have not been calculated due to closure time for investigations.

Increase in Number of Items Tracked in NTS

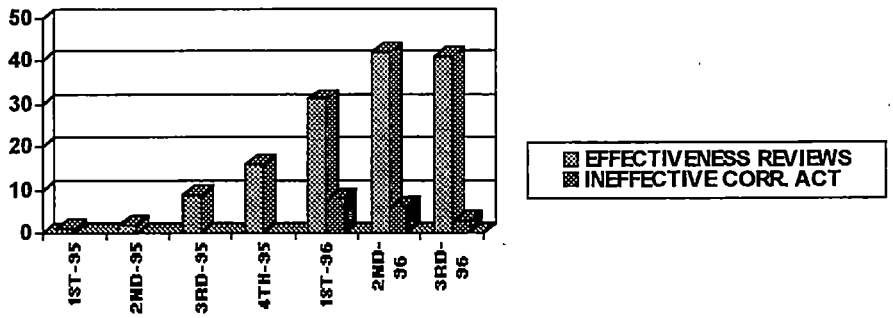
The number of open commitments tracked in NTS has increased significantly. Since January 1995, 520 to 1223 at the end of October 1996. This increase is primarily due to NTS becoming the sole database for the tracking of commitments at the site. The improvement also stems from SQV ensuring unresolved issues relating to PIFs are tracked in NTS until proper closure.

Establishment of Corrective Actions Review Board

In July 1995, a Corrective Actions Review Board (CARB) was implemented at Dresden to review Level 1, 2, and 3 Root Cause Reports. The CARB consists of department head level individuals from Maintenance, Engineering, Services, Operations, and Radiation Protection. These individuals review root cause reports to ensure that the correct primary root cause has been determined, and that the corrective actions will prevent recurrence of the event in the future. Since its was implemented, CARB has reviewed over 600 Root Cause Reports.

Growth in Effectiveness Reviews

Effectiveness reviews are used to determine whether Corrective Actions are effective. The number of effectiveness reviews has increased dramatically since the beginning of 1995. The station attributes the increase in the number of effectiveness reviews to more aggressive action by SQV and CARB in assigning effectiveness reviews for Corrective Actions.



As more effectiveness reviews have been performed, ineffective Corrective Actions have dropped steadily.

4.8.3 SALP

In the past, the processes for problems identification and corrective actions implementation were viewed as weak. In the last SALP report dated April 28, 1995, it was noted that some improvement in this area was made. During the recent NRC Independent Safety Inspection (ISI), it was noted that the identification of plant problems via the PIF process shows overall improvement. Over 5700 PIFs were written in 1995 and over 6700 were generated in 1996. Numerous examples of internal assessments and PIFs demonstrated a pattern of increasing sensitivity of the station personnel to this area, however, areas still requiring improvement were noted, particularly in the Design Engineering organization. Improvement efforts to address this concern are in progress.

4.8.4 Joint Utility Management Audit (JUMA)

A Joint Utility Management Audit (JUMA) performed at Dresden Station in 1996 included an assessment of the Site Quality Verification (SQV) organization. In general, the review team concluded that the SQV program is effectively implemented at Dresden. No significant or programmatic deficiencies were identified; however, opportunities for improvement have been identified, and several improvement initiatives have been undertaken.

4.9 Conclusion

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

5.0 Action (e) 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

ComEd concludes that there is reasonable assurance that the current Dresden Station processes and programs are effective overall in maintaining consistency between the plant's configuration and operation and its design bases. In addition, implementation of these processes and programs have been improved through increased emphasis on timely resolution of issues. Further confirmatory activities and initiatives to improve design basis information are also planned as a result of the NRC Independent Safety Inspection (ISI) in 1996.

As described in Action (a), the station has a complete set of processes and programs that are designed and implemented consistent with industry standards. These processes and programs are specifically designed to assure consistency between the plant's configuration and its design bases. As described in Actions (b) and (c), there is reasonable assurance that Dresden Station's operating, maintenance, and testing procedures substantially reflect and implement the station's design bases, and that the configuration and performance of SSCs are substantially consistent with the design bases. As described in Action (d), identified deficiencies are addressed through the Integrated Reporting Program.

ComEd's overall conclusion for Dresden Station is tempered by the knowledge that some of the technical information behind the design and licensing bases is not always readily available. Dresden's original design was performed by the Nuclear Steam Supply System (NSSS) vendor and Architect Engineer (AE) using engineering standards and design control techniques developed in the 1950s and 1960s. The 1996 NRC ISI verified that some design basis calculations either were not readily available, or in some cases, did not exist. Despite this limitation, for the reasons detailed in this 10 CFR 50.54(f) response, confidence does exist that the station is in substantial conformance with its design and licensing bases.

ComEd's conclusion is based upon a cross-cut of information addressed in Actions (a) through (d). The following five elements of an effective program are satisfied: 1) consistency with design bases at the time of licensing; 2) controls in programs and processes have been implemented since licensing to assure that consistency with the design bases is maintained; 3) improvements to the availability and adequacy of design basis information, and improvements to programs and processes to control changes to them; 4) verification of consistency between plant configuration and design bases through self-assessments, NRC inspections and third-party reviews; and 5) continuation of activities that assure ongoing consistency between the plant and its design bases.

Further reinforcement for these conclusions is provided by management's ongoing communication of its expectations, its policy of providing continuing training for its workers, its sensitizing of the work force to the importance of knowing and understanding the plant's design bases, and its insistence on holding individuals accountable for preserving consistency between

the plant and its design bases. Generally applicable management initiatives encourage the identification and correction of off-normal conditions, including deviations from design bases. Examples of this include extensive configuration management program enhancements in 1995 and 1996 and the November 1996 stop-work-order by the Dresden SQV organization when it became apparent that parts evaluations for balance-of-plant replacement parts were not adequately preserving the plant's design bases.

Finally, the station concludes that the comprehensiveness of the process for developing this response to the NRC's 10 CFR 50.54(f) request, itself provides additional support for the conclusion that plant processes and programs are adequate to assure that the physical configuration of the station is consistent with its design bases. In preparing this response, the site's configuration management processes were extensively reviewed, as well as the results of past ComEd, NRC and third party reviews of these processes. The result is an integrated overview of the site's configuration management program that provides a basis for concluding that there is reasonable assurance that the plant is maintained and operated substantially consistent with its design bases.

5.2 Initial Quality and Accessibility of Design Basis Information

At the time Dresden Station was licensed by the NRC, ComEd demonstrated that there was reasonable assurance that the station's configuration was consistent with its design and licensing bases and that the station's processes and procedures should enable the station to be operated consistent with its design bases. The quality and accessibility of design basis information available to Dresden Station at that time was consistent with then current regulatory expectations.

The original plant startup testing, maintenance and operating procedures were prepared by the NSSS vendor, Architect Engineer and ComEd prior to startup. The startup testing demonstrated that the plant performed as required by the design and licensing bases.

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

From the time of its original construction, Dresden Station has adopted and implemented configuration control processes designed to provide reasonable assurance that changes to the plant's configuration and procedures maintain the plant's consistency with its design bases. The current processes have been described in Action (a). Major enhancements (some of which eliminated observed deficiencies) have been described in Actions (b) and (c). The ability of the corrective action program to identify deficiencies and result in effective improvements has been described in Action (d). These processes, through their integrated performance, have progressively reduced the significance of deficiencies over time and have enabled Dresden Station to identify and correct consequences of those deficiencies.

The effectiveness of the current processes and programs in maintaining the plant consistent with its design bases stems from both their structure and their implementation. With regard to their structure, this response explains how the procedures that implement the processes and programs

are formalized and incorporate cross-functional reviews which provide checks and balances. For example, any proposed procedure change or new procedure that could impact safety is subject to an evaluation under 10 CFR 50.59 and additional reviews by the Onsite Review Group, PORC, and the Station Manager, who bring different perspectives to the review process. Implementation of the procedures is subject to the station's generally applicable expectations for quality control and self-assessment, to continuous management oversight, and to requirements that involved individuals have appropriate training and experience.

Moreover, the effectiveness of these programs and processes has been demonstrated repeatedly by ComEd audits, reviews and assessments, NRC inspections, and third party assessments. Where deficiencies have been identified, corrections have been made.

5.4 Enhancements to Documentation Availability and Configuration Control Programs and Processes

As discussed in Appendix I, during the 1970s through 1990, the ComEd design engineering organization was a corporate centered project management group. It was centrally located and provided project management services for each of the ComEd operating stations. When station modifications were performed, or other significant engineering services were required, the NSSS vendor or an AE was contracted to perform the engineering function. Project management functions were performed by the corporate engineering department, and a Station cognizant engineer was assigned to assist in implementation. Only that portion of the design basis information which was required for implementation was delivered to the station. This would have consisted of reports, drawings, procedures, databases, etc. Underlying calculations, specifications, etc., were retained by the NSSS vendor or AE. Sargent and Lundy was contracted to maintain station drawings and databases. This service included, not only changes originating at Sargent and Lundy, but also coordination of changes originating at the NSSS vendor or other AEs.

This was the ComEd model program through 1986 when the NRC performed the Dresden Safety System Outage Modification Inspection (SSOMI). This inspection is discussed in Action (c). As a result of the Dresden SSOMI, significant improvements were made in the modification process. In the early 1990s, the corporate engineering organization began evolving to a more decentralized model. Site Engineering Departments were established at each site. These departments were still "project management" type organizations; however, better control over design basis information was established. During this time period, Dresden began taking increased ownership of design basis information. For example, drawing revisions and control were performed onsite.

In 1994, the current design engineering model began to evolve. This model encompasses the concept of complete control over Dresden design and licensing basis information, and limited design independence, including: 1) hiring skilled engineers from Architect Engineers and other utilities; 2) developing standard Nuclear Engineering Procedures (NEPs); 3) developing and improving technical and procedural training programs; 4) transferring drawings and calculations into ComEd control; 5) gaining experience within the company to perform modifications, rather

than oversee (project manage) them; and 6) performing safety evaluations and UFSAR updates inhouse.

In addition to this change of focus regarding engineering design, since the mid 1980s, Dresden has taken a number of significant steps to improve the quality, availability and usability of design basis information. These actions include:

- Creation of 23 Design Bases Documents
- Rebaselined UFSAR
- Procedure Upgrade Program
- Onsite Control of Station Drawings
- Plant Labeling Program
- Upgrades to Design Basis Databases, e.g., Master Equipment List, Instrument Data Sheets, Fuse List, and many others
- Consolidated primary design data into the Electronic Work Control System (EWCS)
- Procedure Writer's Guide Development and a Rewrite of All Station Procedures to that Writer's Guide
- Reduction in the quantity of open Design Changes and other Configuration Management Backlog Reduction Efforts
- Reduction in the Number of Temporary Alterations
- Refinement of the AC and DC Electrical Load Monitoring Systems

These initiatives have improved the design basis documentation, provided additional reference information for future modifications and provided controlled database information which is updated as part of the modifications.

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

Routine plant activities, audits, and inspections, and Dresden's responses, are an important element of maintaining conformance with the design bases over the operating life of the plant. Some of these activities have been discussed in response to Actions (a) through (d). The more significant efforts are discussed below.

5.5.1 Verifications As Part of Normal Plant Activities

As described in the response to Action (c), Dresden 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 (IST) programs required by ASME Section XI and 10 CFR 50.55(a). As described in the Appendix II and the response to Action (c), design and configuration control processes require rigorous post maintenance and modification testing. Finally, as described in the responses to Actions (b) and (c), performance of the plant as expected in response to normal plant evaluations, and to equipment failures and transients, provides additional confirmation that plant procedure and SSC configuration and performance have been maintained consistent with the design bases.

5.5.2 Vertical Slice System Functional Assessments

Many important Dresden safety systems have received one or more SSFI type inspections. The inspections have either been performed by the ComEd line organization in conjunction with SQV and third party industry experts, or by the NRC. These inspections are:

Year	System	Organization
1987	Emergency Diesel Generators	ComEd
1988	High Pressure Coolant Injection	ComEd
1991	Electrical Distribution System Functional Inspection	NRC
1993	Service Water System Operational Performance Inspection	NRC
1996	Standby Gas Treatment Design Basis Document Validation	ComEd
1996	Low Pressure Coolant Injection	ComEd
1996	Containment Cooling Service Water	ComEd
1996	Core Spray	NRC ISI
1996	125 Volt DC	NRC ISI
1996	High Pressure Coolant Injection	NRC ISI

Historically, these reviews have revealed significant deficiencies in the control and availability of design basis information at the station. As a result, specific configuration problems have been addressed, programs have been enhanced, and design basis information has been improved. While efforts in these areas continue, this record of assessment and corrective action provides a substantial basis for the conclusion that the systems are currently safely configured. Few recent issues have been identified that led to determinations of system inoperability.

5.5.3 Latent Material Condition and Material Condition Improvement Initiative

In 1995 a team of third party industry experts performed an extensive review of key Dresden systems to determine, based on industry experience, material conditions issues which had not yet manifested themselves in open plant performance. The identified issues were compiled and prioritized. On going system improvement plans were developed in 1996 to address the more significant issues. These prioritized issues are evaluated in the station's business planning process.

5.5.4 Configuration Management Self-Assessments and Backlog Reduction Efforts

In 1995, a number of significant investigations were conducted related to Dresden's configuration management programs. These investigations revealed a number of programmatic configuration management issues as well as a large number of backlog issues. Corrective actions to address the programmatic issues were implemented, which have resulted in significant improvements to Dresden configuration management processes. Goals were established and resources assigned to address the backlog issues. Monthly progress was tracked and all identified backlogs were eliminated or reduced.

5.5.5 1996 NRC Engineering and Technical Support (E&TS) Inspection

The 1996 NRC E&TS inspection was conducted by a team of six NRC and NRC contractor personnel over a four week period. Positive comments were made on the Dresden design change process. Some issues were identified including: problems with post modification testing, minor errors in calculations, and isolated examples of inadequate 10 CFR 50.59 evaluations. In general, it was a positive inspection. No violations of NRC requirements were discovered.

5.5.6 UFSAR Reviews

In 1996, extensive UFSAR reviews were conducted. These reviews included: 1) Reviews of the UFSAR against Dresden Operating Procedures and Surveillances; 2) Detailed reviews of the UFSAR against Core Spray, HPCI, LPCI/CCSW and 125 V DC requirements; and 3) Significant review to support the Technical Specification Upgrade Project. The result of these reviews is significant confidence that the station is in substantial conformance with UFSAR requirements.

5.5.7 Integrated Reporting Process (IRP) and Routine SQV Monitoring

The primary means for problem identification is via normal day to day activities in which the worker or supervisor detects a potential condition adverse to quality and generates a Problem Identification Form (PIF).

Consistent with management's expectations, over 10,000 PIFs have been generated since 1994 and between eighty and ninety percent of all PIFs are initiated early in the quality barrier process by workers or supervisors.

SQV audits have been effective at identifying design basis conformance, and conditions adverse to quality. SQV identified weaknesses in design control, thereby confirming the effectiveness of the audit process in discovering problems which could impact the design control area. Recently, NRC has commented favorably on the effectiveness of the quality verification process for its effectiveness in identifying problems. SQV trends PIF data and reports the results of that effort on a periodic basis. Level three investigation are conducted when trend data indicate a problem in any area.

5.5.8 NRC Independent Safety Inspection

From October to December 1996, the NRC conducted an ISI at Dresden. The 1996 NRC ISI raised significant concerns regarding the accuracy, availability, control and maintenance of design basis information; especially as it related to calculations. These issues are summarized as follows:

- Discrepancies between Design Basis Documents, UFSAR, and other Design Documents;
- Some engineering evaluations were weak, e.g., operability and 10 CFR 50.59 evaluations;
- Engineering failed to resolve some safety issues in a timely manner;
- Engineers are not effectively using the Integrated Reporting Process (IRP);

- Calculations not readily available;
- Calculations poorly indexed and cross-referenced;
- Calculations not updated when related calculation revised or voided;
- No supporting calculation for important design basis information;
- Conflicting calculations for the same equipment; and
- Calculation errors.

During the 1996 NRC ISI, numerous operability evaluations were performed. In only one case was a system required to be declared "Inoperable" -- the Drywell Inerting System was briefly declared to be inoperable while hydrogen generation calculations (based on original nuclear fuel) were revised for current nuclear fuel; the revised calculation showed that the system met current licensing requirements and operability was restored. In all other cases, a basis for system operability was established. Although significant issues on the control of design basis information were raised, in all cases, system operability was established.

5.5.9 Review of Risk-Significant Systems

In late 1996, following the NRC ISI, a review was conducted of twelve risk significant systems at Dresden Station. This review included an evaluation of current surveillance and acceptance criteria relative to design functions identified in the UFSAR. The review also encompassed backlogs, Licensee Event Reports, and Nuclear Tracking System commitments for each system. In each case it was determined that the surveillance results demonstrate that the equipment operates as expected and will perform the intended safety function.

5.6 Continuation of Design Conformance Activities

Dresden Station was designed and constructed in the 1960s as a "turn-key" plant. The NSSS vendor retained possession and control of all NSSS design basis information. The AE retained possession and control of all AE design basis information. When design basis questions arose, the NSSS vendor or AE was contacted, and the required information was provided.

In recognition of issues raised by the 1996 Dresden ISI and discovered in preparation of this response to NRC's 10 CFR 50.54(f) letter, ComEd will continue and expand its program to increase the accessibility and retrievability of existing design basis information (primarily calculations). In addition, for the most risk significant systems, ComEd will review, validate and, where necessary, reconstitute design basis information.

Additionally, Dresden Station will continue with its programs for assuring consistency between the configuration of the plant and its design and licensing bases are ingrained in the station's work ethic. A program to enhance the questioning attitude of engineers in identification and resolution of design basis problems will be developed. One feature of this program will be to assure that engineering personnel make greater use of the Integrated Reporting Process (IRP) to assist in design basis problem resolution. The SQV will continue (and increase) efforts to monitor issues related to design basis conformity.

The calculation improvement and design basis validation/reconstitution programs are discussed below.

5.6.1 Improvements in the Accessibility and Retrievability of Calculations

The original scope of "Design Calculation Turnover" was to index and obtain "high use calculations." High use calculations was that subset of the total calculation program which were frequently referenced or revised. During the 1996 NRC ISI, it was realized that there were many other calculations which were important to the design and licensing bases which had not been obtained in the original calculation turnover project. In response, Dresden has developed a program to:

- Obtain indices of the known population of Architect Engineering generated calculations and to transfer them into the common calculation index database (Electronic Work Control System (EWCS));

- Take possession and control of the calculations which are important to (or form the basis of) the design and licensing bases of the plant; and

- Develop augmented indexing for the calculations which are important to the design and licensing bases of the plant.

5.6.2 Design Review and Reconstitution Program Plan

In the late 1980s, ComEd began to assess the quality and availability of the design information for each of its plants. The extent and level of detail of the available information varied considerably across the ComEd plants. There was a marked difference between the older plants (Dresden, Quad Cities, Zion) and the newer plants (LaSalle, Byron, and Braidwood). The former were designed in the 1960s prior to 10 CFR 50 Appendices A and B, with original FSARs of approximately three to five volumes. The latter were licensed in the 1980s in accordance with Standard Review Plans, with FSARs of 12 to 18 volumes.

At Dresden, design input requirements and summaries of design analyses have been assembled in DBDs to provide the rationale for the information documented in design output documents. NUMARC 90-12 was used as a guidance document in this effort. (See Appendix I for definition of design inputs, design analyses, and design outputs). Twenty-three, (twenty system and three topical), DBDs were prepared for Dresden Station during the period 1991 through 1996; one DBD has been validated.

ComEd has developed plans, as documented in the January 30, 1997 T. J. Maiman letter to A. B. Beach, to improve and expand design basis information. This plan has four elements:

- Generation of 10 additional DBDs;
- Validation of all DBDs;

- Reconstitution of design basis calculations, as necessary, for twelve risk-significant systems; and
- Ongoing UFSAR validation efforts.

Additional measures related to the quality of and access to design basis information were described in the November 12, 1996 T. J. Maiman letter to A. B. Beach.

5.7 Conclusion

Dresden Station recognizes historic weaknesses in certain programs and processes, as well as weaknesses in the availability of certain calculations supporting the design bases. Nonetheless, based on substantial information described in this response, there is reasonable assurance of substantial consistency between the plant and its design and licensing bases. ComEd has taken actions to address identified weakness, and significant past initiatives, reviews, and verifications provide the reasonable assurance that the plant can safely operate.

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

1.0 Role of ComEd Engineering in Design Bases Management

The Station Engineering Organization plays a significant role in controlling, maintaining, and 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 to a need to further transition the role of Engineering to one with a more active focus directly at the station. Transition of major responsibilities to Engineering and the role of Corporate versus Site Engineering in assuring design bases conformance 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 combined engineering and construction team with Engineering producing design and analysis by predominantly managing architect engineering (AE) contracts from the General Office (essentially a Category 3 organization as described in Section 2.2.3 of NUREG 1397). Problem solving and system engineering functions were organized under a technical staff that reported on site to the Station Manager. In 1990 small engineering groups were established on-site to provide a closer presence to the customer base. In late 1992, the nuclear organization was changed to establish authority and accountability on-site under a Site Vice President.

Multiple architect engineers were used; but a common approach was assured by use of an AE guidebook. This guidebook formalized the interfaces and communication channels between ComEd and the AE. During this period, 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 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. This review identified eleven strategic issues and targeted recommendations to deal with those issues. Key amongst them was the understanding and "owning" of the design. ComEd clearly had to become more knowledgeable in the design, license, and operating bases of the plants. We needed to be in a stronger position to control the design configuration and be proactive in any matter that requires design information to resolve. The TENERA Report provided recommendations regarding access to and control of design information, and suggested that the first priority should be assigned to efforts required to take ownership of the design and develop in-house capability. It also included a recommendation for development and implementation of a plan for consolidation of design information under ComEd control.

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

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

While we have essentially achieved Category 2 (NUREG 1397) status, our goal is to reach Category 1. We feel our success lies in qualified people, common and controlled processes (including Corrective Action), and being our own worst critic. Our commitment to conducting rigorous safety system functional inspections will provide "Reasonable Assurance" that we are maintaining the design bases.

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. Information transfer prevents problems by sharing information, assists in problem solving, provides clear knowledge of the design bases, shares information for design modifications of similar components or for similar systems from site to site, and shares the results of assessment and oversight activities. The key information transfer vehicles that have been used are a daily engineering phone call, a Tech Alert program, Corporate Engineering oversight of station activities, the Engineering Managers Team meeting, and Engineering Peer Groups.

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

Corporate Engineering Oversight Role - The Chief's staffs periodically review 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

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

At the corporate level, there is a Chief Engineer, Configuration Management, reporting to the Engineering Vice President. The Chief is accountable for setting policy for configuration management and implementing the policy through a series of common processes and procedures. These common processes are documented in a set of Nuclear Engineering Procedures (NEPs) used 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 supervises the close-out of the design changes to ensure all controlled documentation (with the exception of procedures) and databases are updated in a timely manner. Procedure update is the accountability of the procedure update group, which is part of the Operations Department.

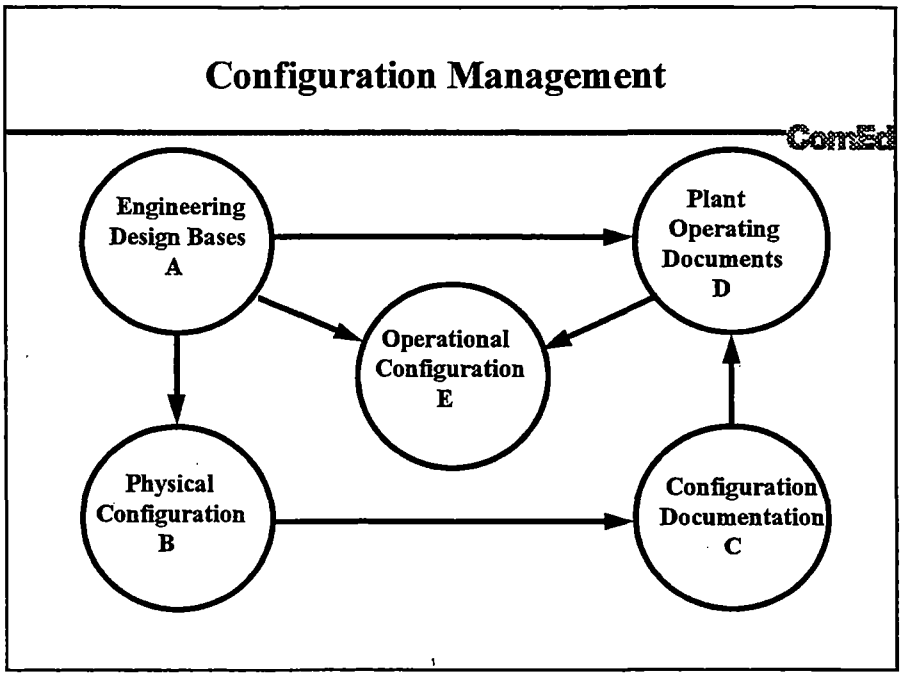
1.4 Configuration Management Model

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).

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



2.0 SELF 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 Engineering Assurance role, were established to provide an extra level of reviews of products prepared by the Site Engineering Organizations in recognition of their new and expanded role onsite.

2.1 Corporate Engineering Assurance

The Corporate Engineering Assurance Function is part of the Configuration Management organization. The role of this group is to provide technical assurance that the work performed by Architect Engineers and other contractors is in conformance with ComEd's Nuclear Engineering

Procedures and the QA Manual. This is accomplished through periodic audits of the AEs, generally in a teaming arrangement with the Quality Assurance Department.

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

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

2.2 Site Engineering Assurance

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

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

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

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

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

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 and concurrent 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 operability assessments, Safety Evaluations, and Licensing Event Reports. The Offsite Review for each document requires two participants and an approval signature. As reviews are completed, they are transmitted to the Sites. Reviews may have comments and recommendations or actions assigned based on the completeness of information contained in the document.

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

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

2.4 Role of Corporate 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.

Appendix II Design Control and Configuration Control Processes

Background

This appendix summarizes the major processes used at Dresden Station to control the plant's design bases and configuration, i.e., maintaining the physical plant consistent with the documented plant and with design bases. These processes are designed to ensure the design bases of the plant are maintained or modified as changes are made to the plant as a result of modifications, repairs, or equipment lineup changes. This appendix supports the description of configuration control and design control processes as required for Action (a) of the 10 CFR 50.54(f) response

Matrix of Appendix II Process

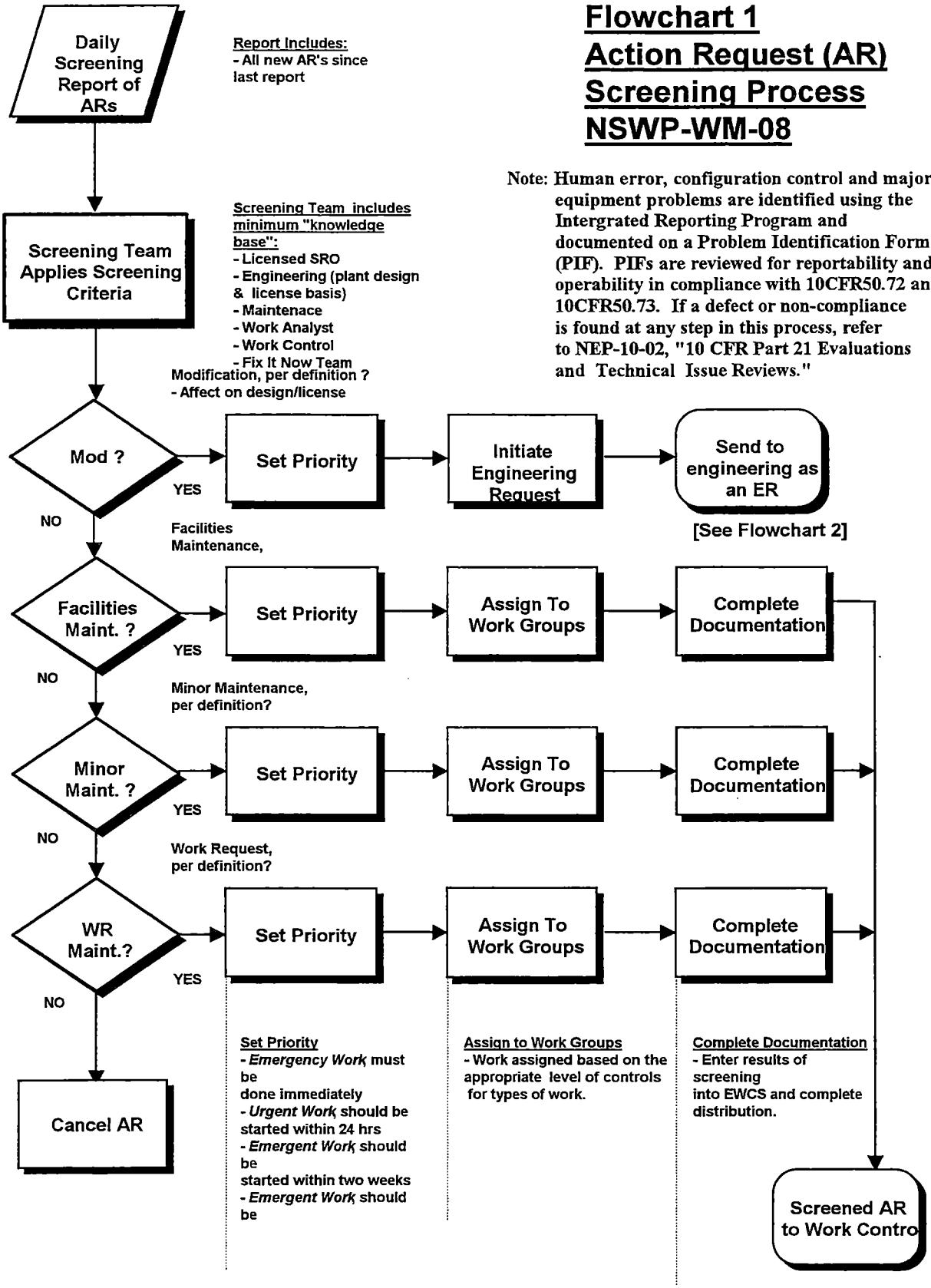
Process Number	Process Description	Procedure Reference	Implements Regulatory Requirement		Configuration Management Model Linkages (note 1)						
			50.59	50.71(e)	A/A	A/B	B/C	C/D	A/D	A/E	D/E
1	Action Request (AR) Screening Process	NSWP-WM-08				X	X			X	X
2	Overview of Design Control Process				X	X	X	X	X		
3	Design/Document Change Processes Roadmap	NEP-04-00			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) Process	DAP 05-08	X			X	X	X	X		
7	Document Change Requests (DCRs)	DAP 21-11	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 Process	DAP 11-11	X	X	X	X	X	X			
10	Design Basis Document (DBD) Update Process	NEP-17-01				X	X				
11	Engineering Software Development and Revision Process	NEP-20-01			X						
12	Engineering Change Notices (ECNs)	NEP-08-01				X	X	X			
13	Safety Evaluation Process	DAP 10-02	X	X	X						
14	VETIP Processing	NEP-07-04				X	X	X			
15	Configuration Control Using EWCS	NEP-14-01				X	X				
16	DBD Development Process	NEP-17-01				X	X		X		
17	Calculation Process	NEP-12-02			X	X					
18	Operability Determination Process	DAP 07-31			X	X			X		
19	UFSAR Update Process	DAP 21-06	X	X	X	X	X	X	X		
20	Out of Service/Return to Service Process	DAP 03-05								X	X

Notes:

1. A/A Link are processes that affect the design bases only.
2. Applies to Field Change Requests when needed.

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 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.

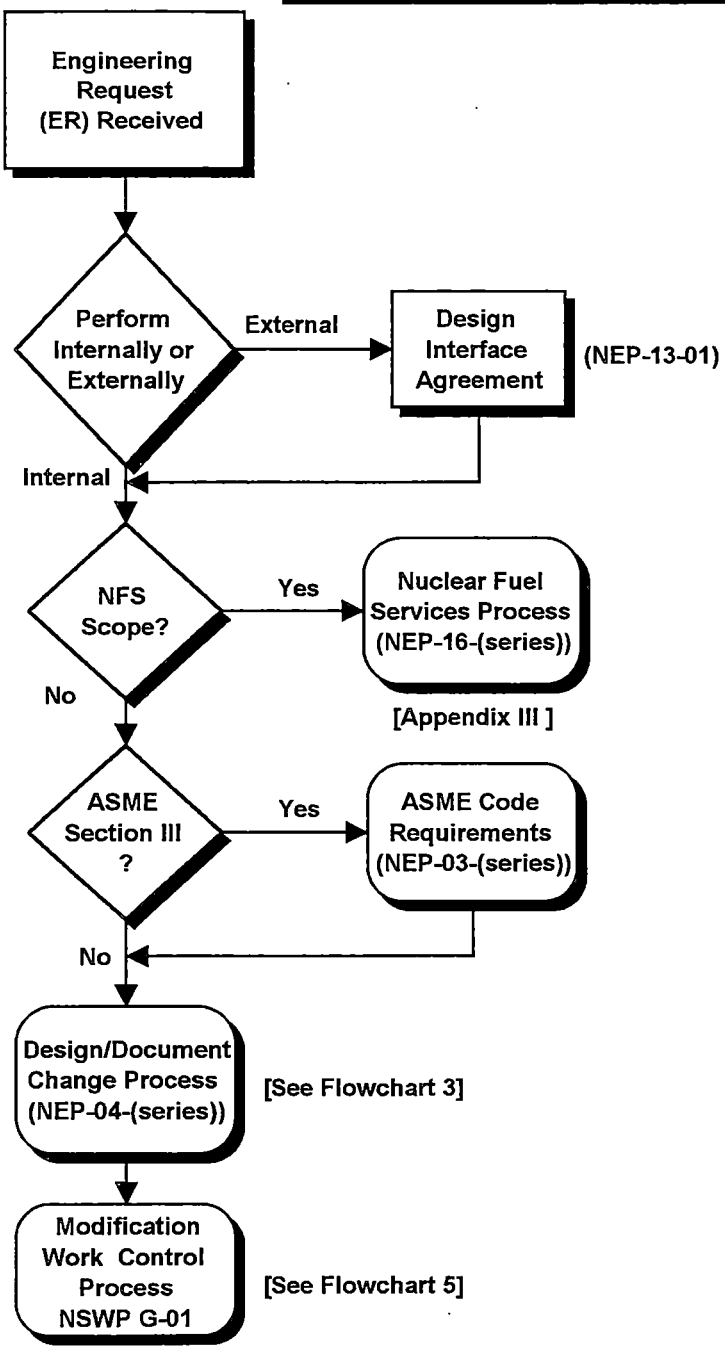
Increased emphasis has also been placed on the definition of Facility Maintenance, Minor Maintenance, and Work Request Maintenance. In each of these types of work, clear boundaries have been provided to maintain the appropriate level of controls. If during the process something requires work to fall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

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

Flowchart 2
Overview of Design Control Process



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

Overview Of 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.

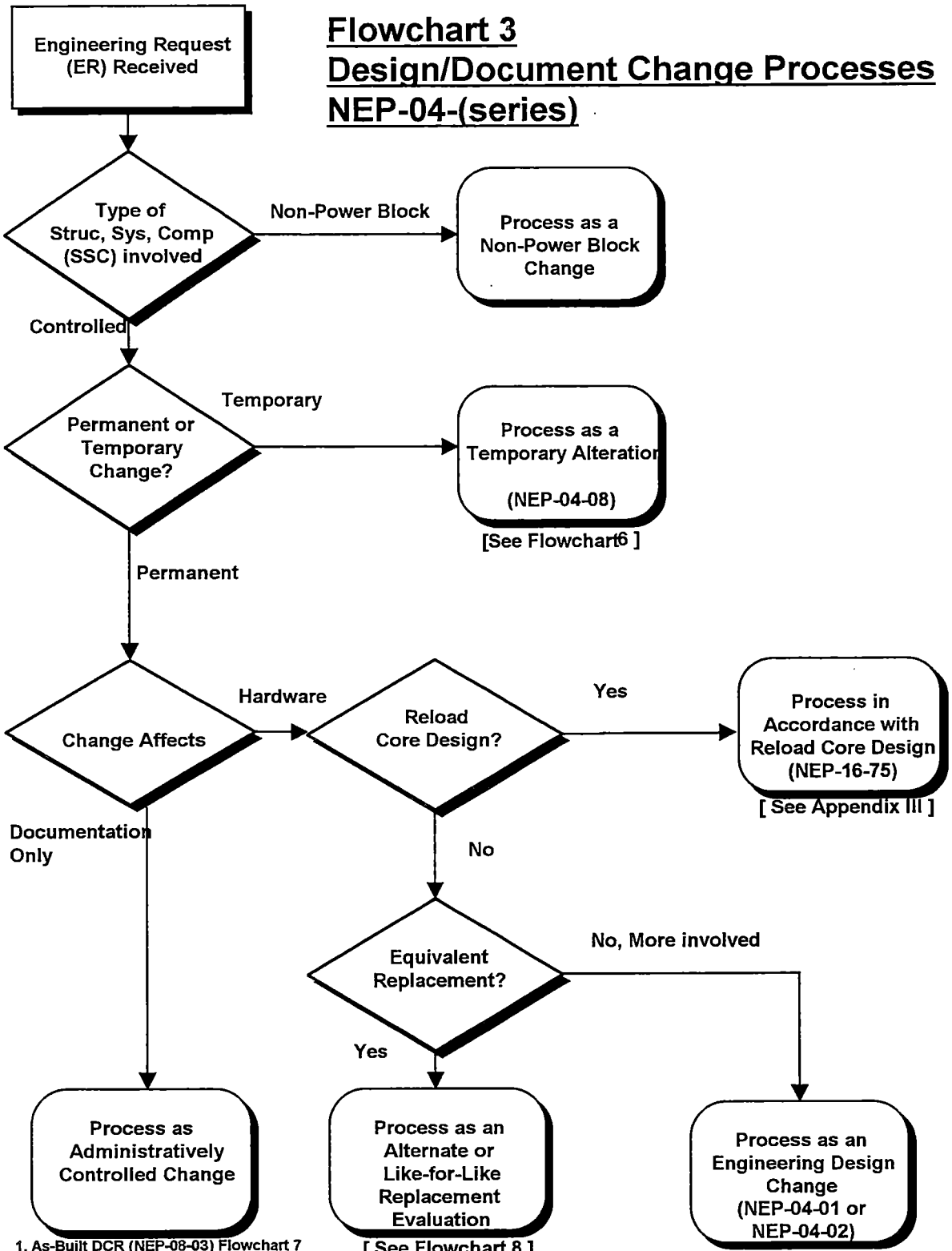
The Design Change Process and the Modification Work Control Process will be described separately in the detailed process descriptions that follow.

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 Performance Improvement 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."

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



1. As-Built DCR (NEP-08-03) Flowchart 7
2. Setpoint Change (NOD-MA.10) Flowchart 9
3. Design Software Revision (NEP-20-01) Flowchart 11
4. UFSAR (Plant Procedure)
5. Design Basis Document (ENC-QE-76.1) Flowchart 10

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

Temporary Alterations - The second decision point determines if the proposed change is permanent or temporary. Temporary Alterations 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.

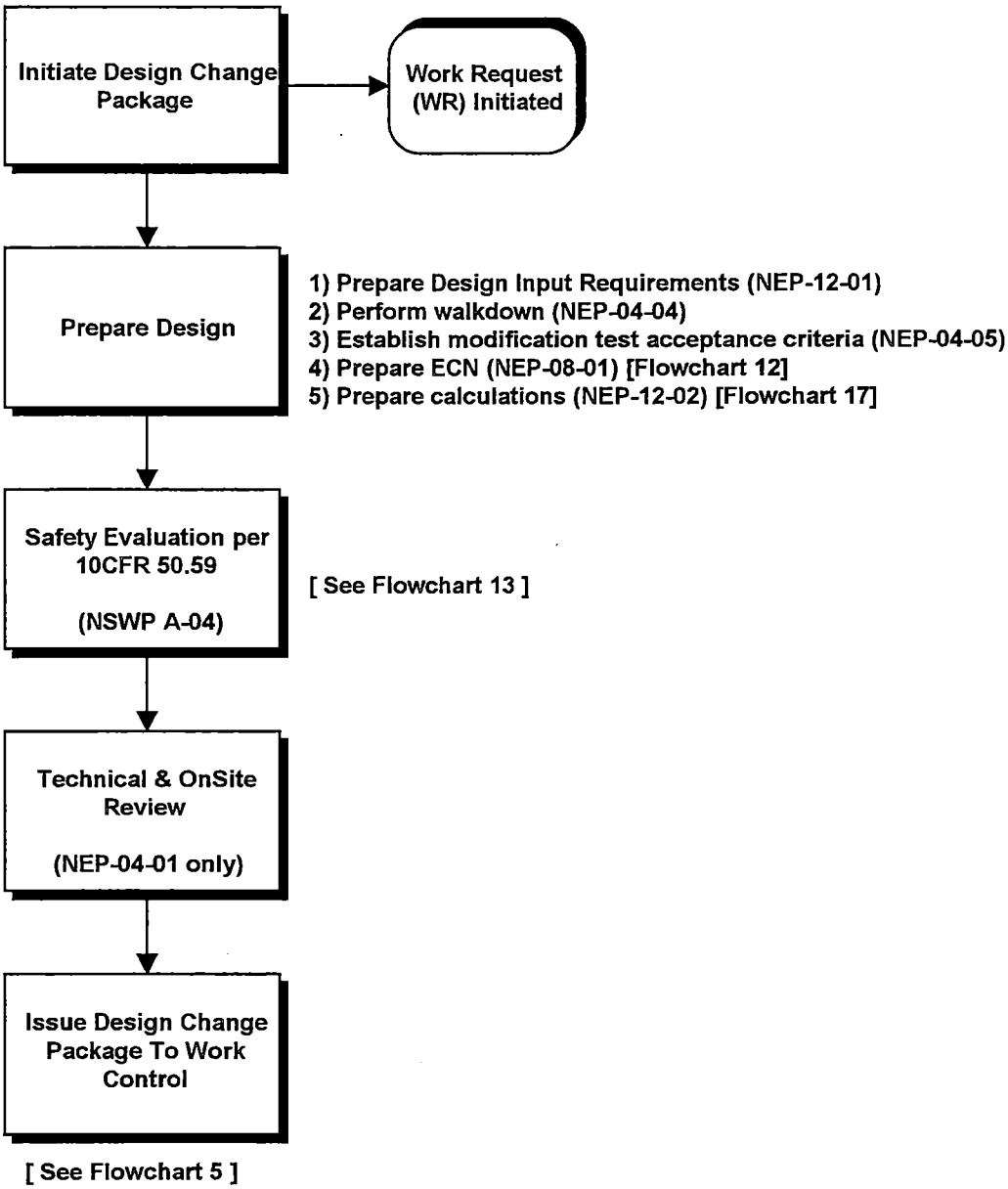
CHECKS AND BALANCES

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, ComEd has established a separate process for handling these "Non-Power Block Changes." This process is currently being used at Dresden. 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 is held. This meeting brings together appropriate Engineering, Operations, Maintenance and Support personnel to review the scope and schedule for the modification, define responsibilities, determine deliverables, review the preliminary design, identify and confirm design inputs, perform a pre-design walkdown and resolve or identify 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."

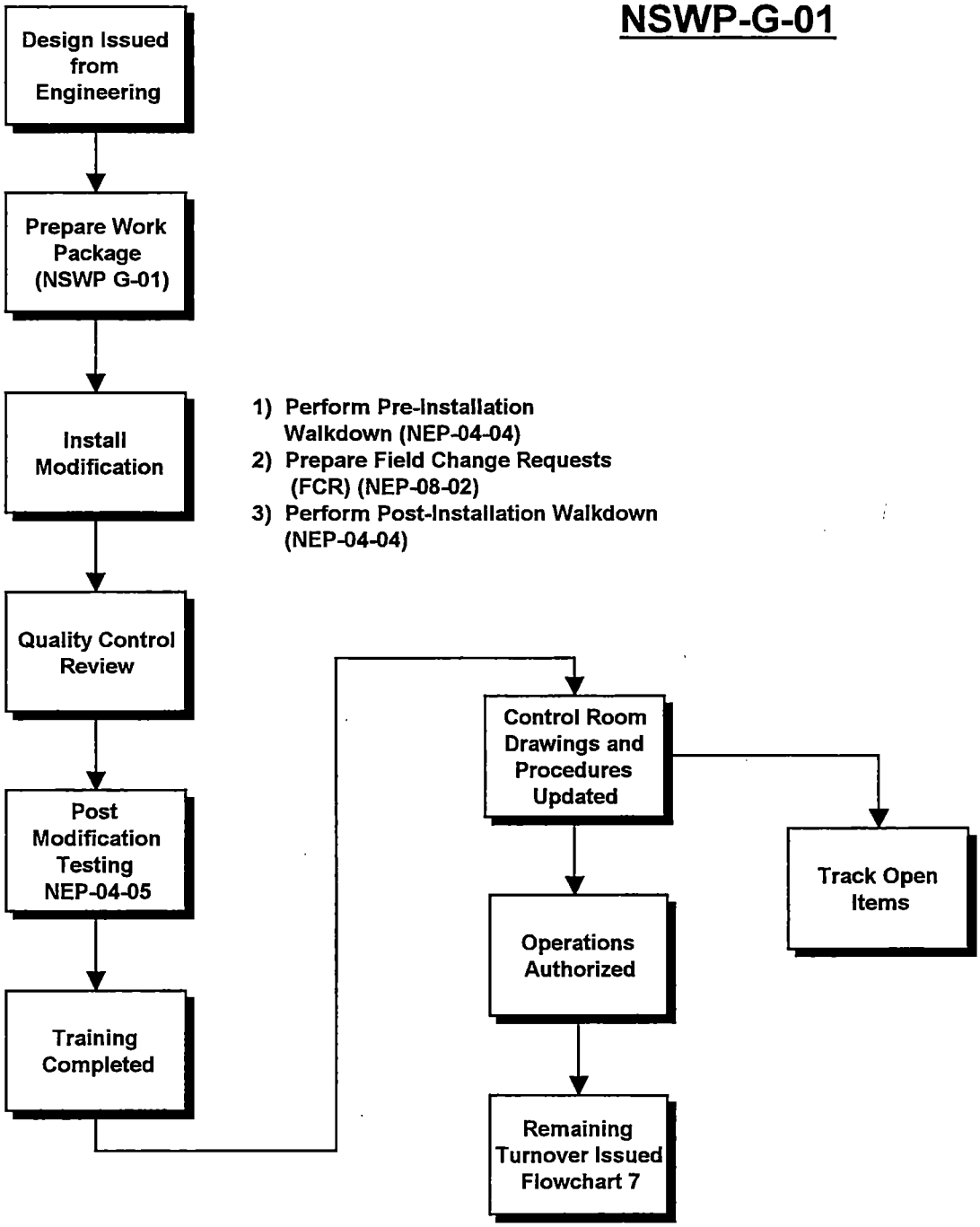
CHECKS AND BALANCES

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. 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.

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. 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.

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

Flowchart 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 issued, a Work Package is prepared (see Action "a", section 1.4) 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

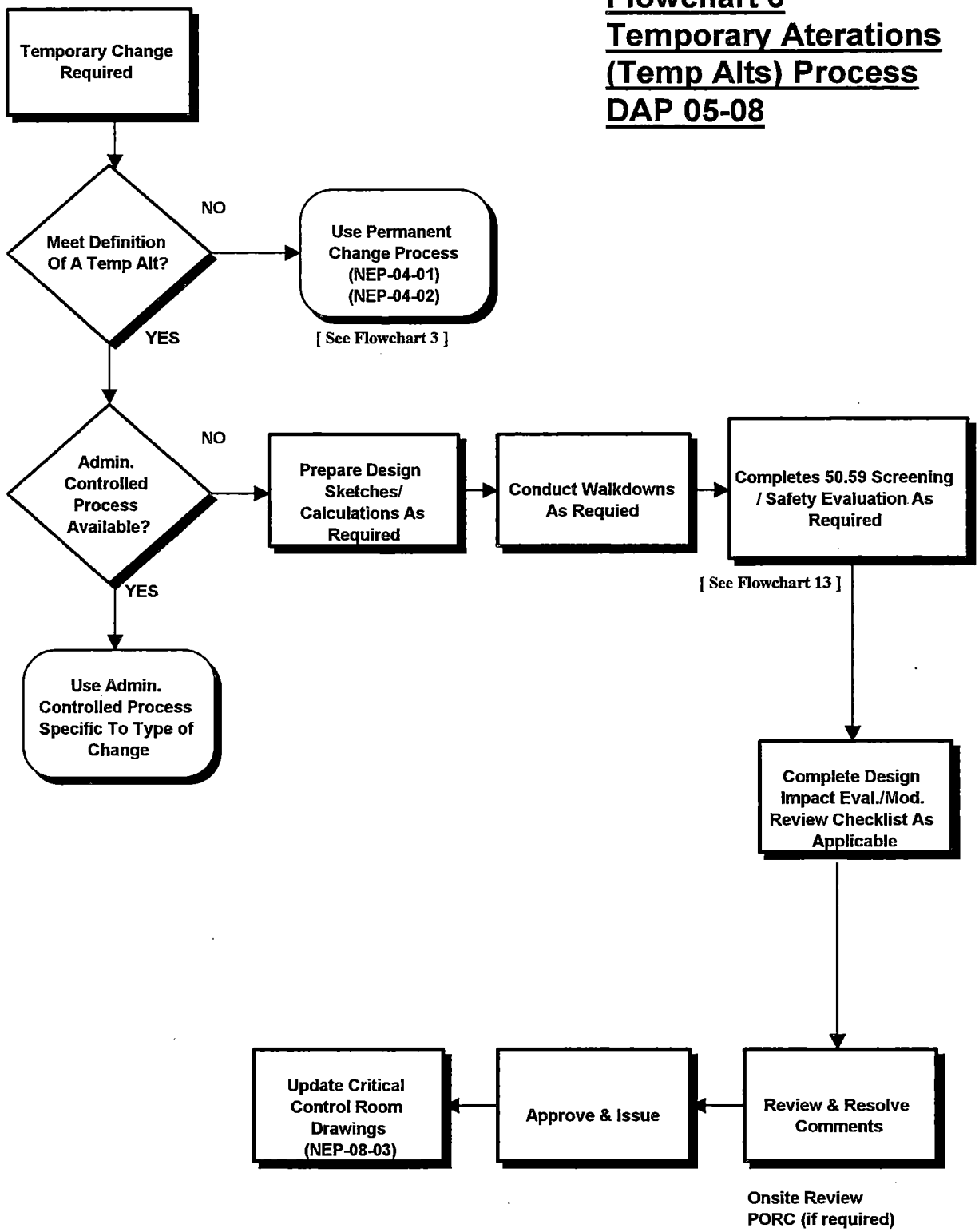
The pre-installation walkdowns provide an opportunity to evaluate the modification against the physical attributes and design considerations of other components located in same area. Any changes required during this evaluation and others required during the installation, are all evaluated through the 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 10 CFR 50.59 Safety Evaluations, if applicable.

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

RECENT/PLANNED IMPROVEMENTS

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

Flowchart 6
Temporary Aterations (Temp Alts) Process
DAP 05-08



Temporary Alterations (Temp Alts) Process

DAP 05-08

PURPOSE

The impact of Temporary Alterations on the plant design and licensing bases documents is controlled through a detailed preparation and review processes. This process is intended to assure that a Temporary Alteration does not degrade plant safety/reliability or unacceptably alter the approved and controlled design configuration.

Initial Review

The initial request for a Temporary Alteration is reviewed by the Program Owner to determine if the change is actually needed and if the change should be a permanent installation. If the change is determined to be permanent, the request is processed as an Exempt Change.

Preparation

Temporary Alterations are prepared by qualified Design Engineering Personnel. During preparation, the Engineer reviews the Temporary Alteration for:

- Possible impact on Transient and LOCA Analysis Input Parameters.
- Impact on drawings, calculations, procedures, etc.
- To ensure that plant safety and reliability is not adversely affected.
- Material and testing considerations.
- To ensure approved design configuration is not unacceptably altered.

10 CFR 50.59 Screening/Safety Evaluation

A 10 CFR 50.59 Screening/Safety Evaluation is performed to provide the basis for determining whether the Temporary Alteration could involve an Unreviewed Safety Question or a change to the Technical Specifications.

- All Temporary Alterations receive, as a minimum, a 10 CFR 50.59 Facility Screening.
- 10 CFR 50.59 Screenings/Evaluations are performed and reviewed by qualified individuals.

Independent Review

An "Independent Review" of the Temporary Alteration is performed by a qualified individual from Design Engineering who is discipline or knowledge related to the particular type of installation. This individual shall have had no influence on inputs or approaches utilized in the design development. Review includes:

- Adequacy of design.
- Verification of adequate design documentation.

Design Engineering Approval

Temporary Alterations are reviewed and approved by the Design Engineering Superintendent or designee.

Onsite Review & Investigative Function (OnSR&IF)

All proposed changes or modifications to plant systems or equipment that affect nuclear safety receive a critical and thorough administrative Onsite Review. Onsite Reviews are performed by at least two individuals who collectively possess background and qualification in the subject matter. The Stations Technical Specifications require that Onsite Review personnel meet the applicable experience requirements of Sections 4.2 and 4.4 of ANSI N18.1-1971, Standard for Selection and Training of Nuclear Power Plant Personnel.

Plant Operations Review Committee (PORC)

Temporary Alterations generally do NOT require a PORC review. However, a PORC review may be requested.

Unit Supervisor Review

The Unit Supervisor reviews Temporary Alterations for the following:

- Placement or removal will not place the Unit in an unsafe condition.
- An Action Request/Work Request is initiated to correct the condition requiring the Temporary Alteration.
- Appropriate signatures and dates are indicated on the applicable forms.
- Location information is specific enough to make identification easy.
- Adequate information is available to Operators to understand the effects of the Temporary Alteration.

Independent Verification

After installation of the Temporary Alteration, an Independent Verification is performed to verify the position or condition of the altered equipment.

CHECKS AND BALANCES

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

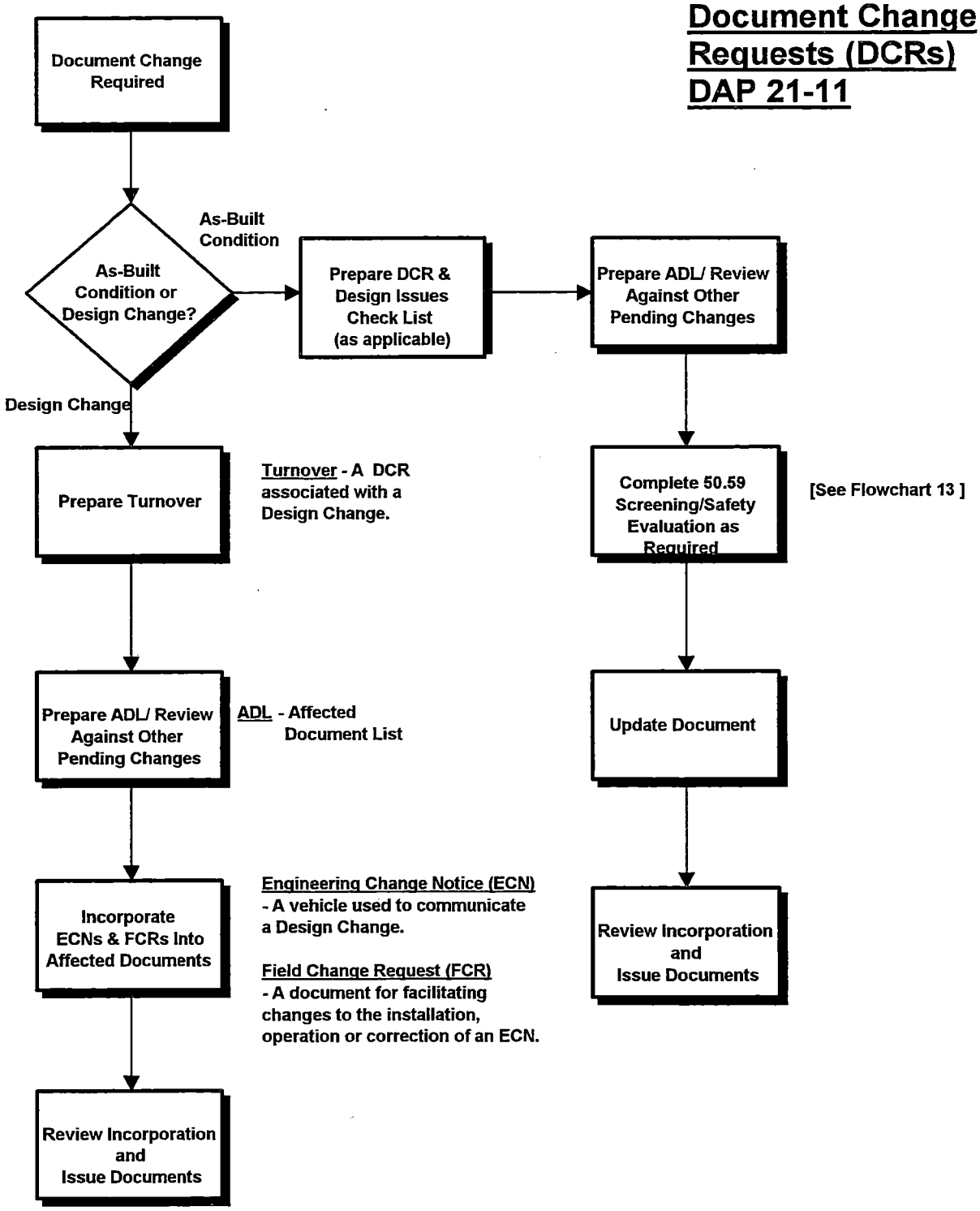
Secondly, On a quarterly basis, the Temporary Alteration Program Owner and applicable Cognizant Engineers perform walkdowns of all installed Temporary Alterations to verify current configuration against original design.

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 Temporary Alteration 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 Temporary Alteration and standardize the process at all six sites. Implementation is planned for early 1997.

Flowchart 7
Document Change Requests (DCRs)
DAP 21-11



Document Change Requests (DCRs)

DAP 21-11

PURPOSE

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

PROCESS DESCRIPTION

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

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

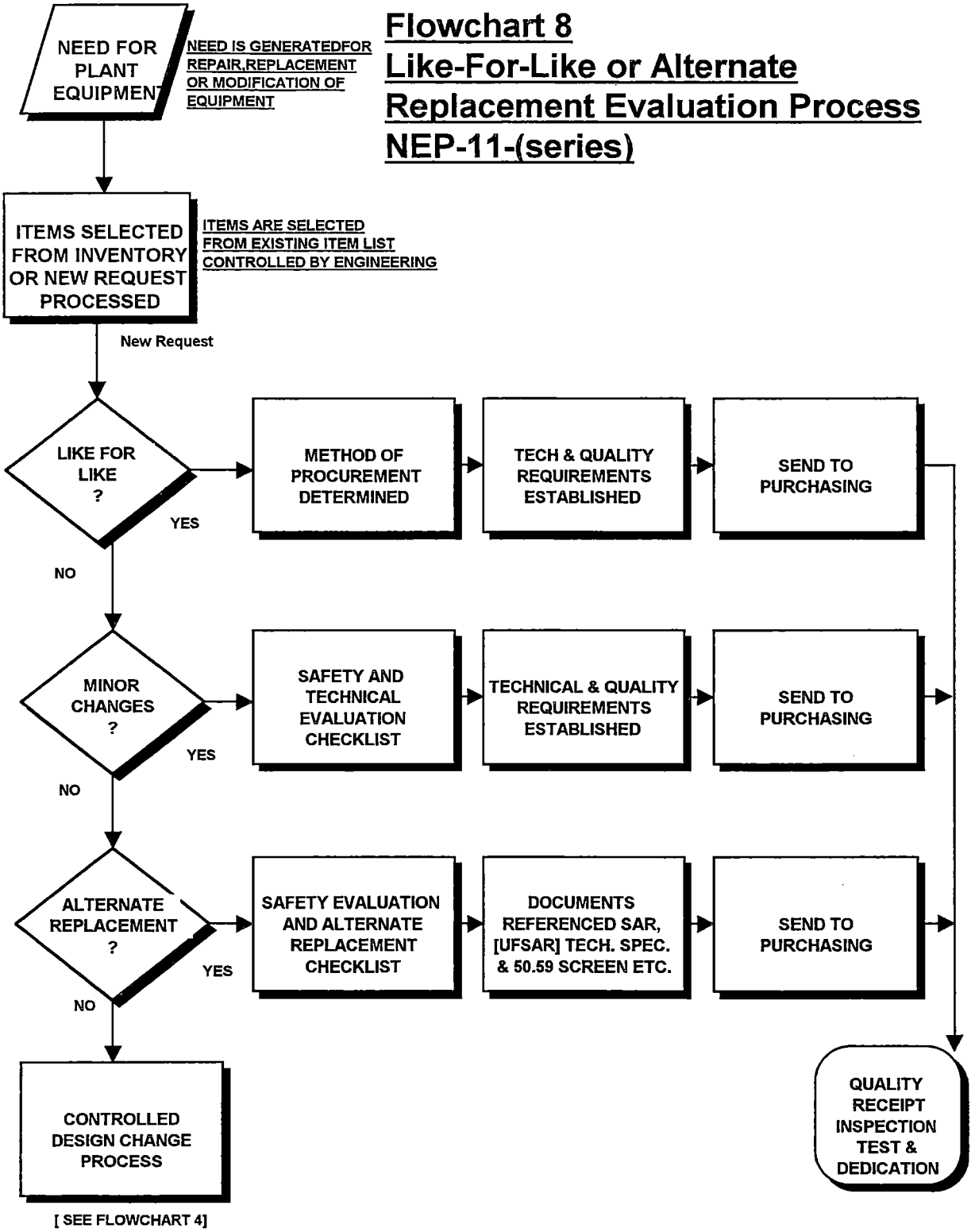
CHECKS AND BALANCES

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

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

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

Flowchart 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 and conferred the title of Good Practice on the material procurement dedication processes.

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

PROCESS DESCRIPTION

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

Should a replacement other than like-for-like [identical] design be required, the process directs the user to the correct procedures for continuation of the process depending on the complexity: Technical Evaluation [NEP-11-01], Alternate Replacement [NEP-11-01], or Modification [NEP-04-01]. The process includes a 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.

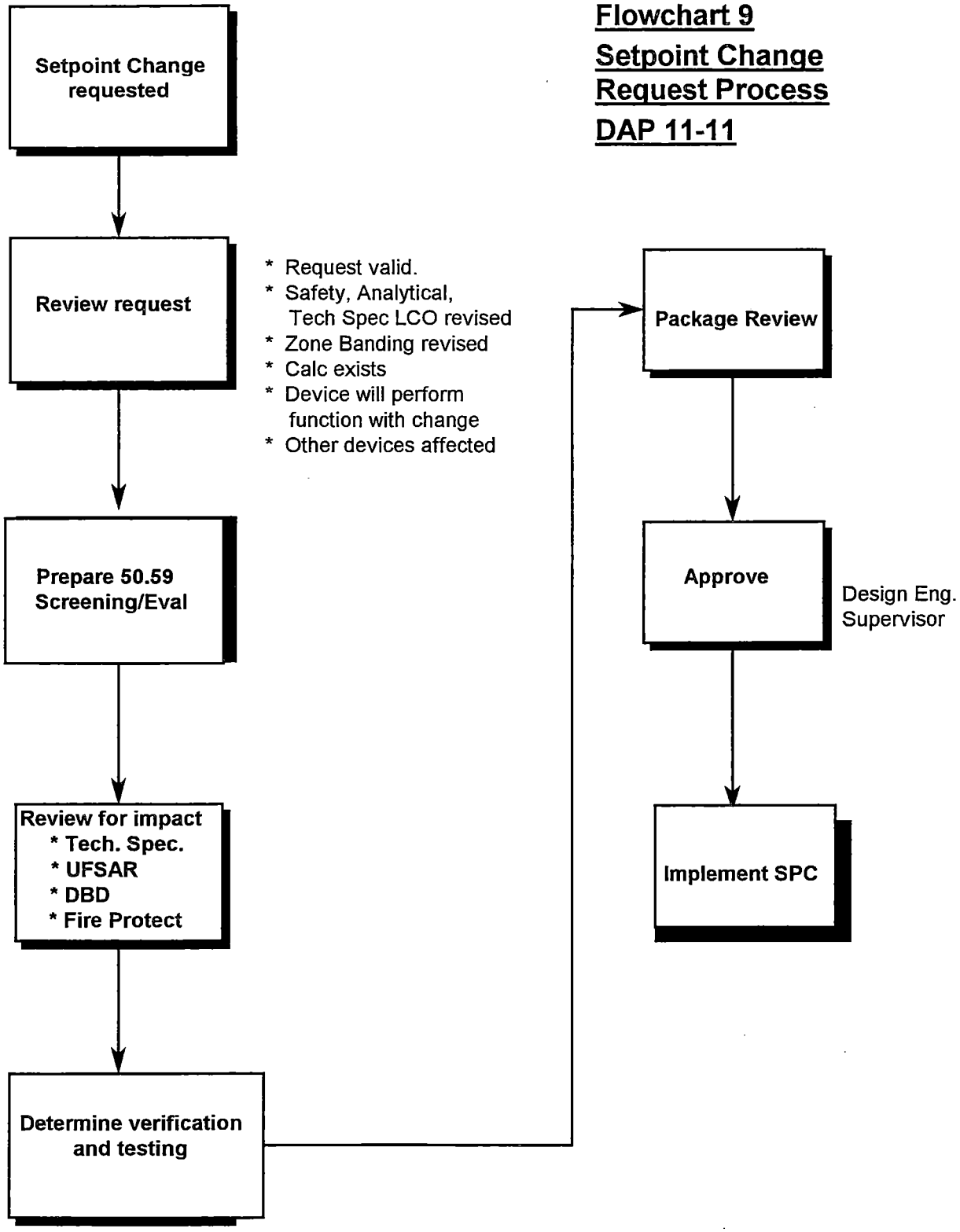
- Application of parts engineering procedures, and process was mandatory for safety related and regulatory related equipment only. Use of procedures and process was optional for non safety equipment.

RECENT/PLANNED IMPROVEMENTS

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 Process
DAP 11-11



Setpoint Change Request Process

DAP 11-11

PURPOSE

The Setpoint Change process is used to change setpoints or scalings when the change is not associated with a design change. Impact of Setpoint Changes to the plants Design Basis and Licensing Documents is controlled through detailed preparation and review processes.

Preparation

Setpoint Changes are prepared by qualified Design Engineering Personnel. During preparation, the Engineer reviews the Setpoint Change for:

- Impact on Technical Specifications.
- Impact to the UFSAR.
- Impact to the Design Basis Documents.
- Impact to the Fire Protection Program.
- Impact on drawings.
- Adequacy of Calculations.
- Possible impact on Transient and LOCA Analysis Input Parameters.
- Required Zone Banding changes.
- Human Factors Engineering changes.

10 CFR 50.59 Screening/Safety Evaluation

A 10 CFR 50.59 Screening/Safety Evaluation is performed to provide the basis for determining whether the Setpoint Change could involve an Unreviewed Safety Question or a change to the Technical Specifications.

- All Setpoint Changes receive, as a minimum, a 10 CFR 50.59 Facility Screening.
- 10 CFR 50.59 Screenings/Evaluations are performed and reviewed by qualified individuals.

Review

Setpoint Changes are reviewed by a qualified individual who is discipline or knowledge related to the particular type of installation. Reviewer's are assigned by the applicable Engineering Supervisor. Review includes:

- Adequacy of design.
- Verification of adequate design documentation.

Design Engineering Approval

Setpoint Changes are reviewed and Approved by the applicable Design Engineering Supervisor.

Onsite Review & Investigative Function (OnSR&IF)

Setpoint Changes which require a 10 CFR 50.59 Safety Evaluation may be "changes or modifications to plant systems or equipment that affect nuclear safety" All proposed changes or modifications to plant systems or equipment that affect nuclear safety receive a critical and thorough administrative Onsite Review.

Onsite Reviews are performed by at least two individuals who collectively possess background and qualification in the subject matter. The Stations Technical Specifications require that Onsite Review personnel meet the applicable experience requirements of Sections 4.2 and 4.4 of ANSI N18.1-1971, Standard for Selection and Training of Nuclear Power Plant Personnel.

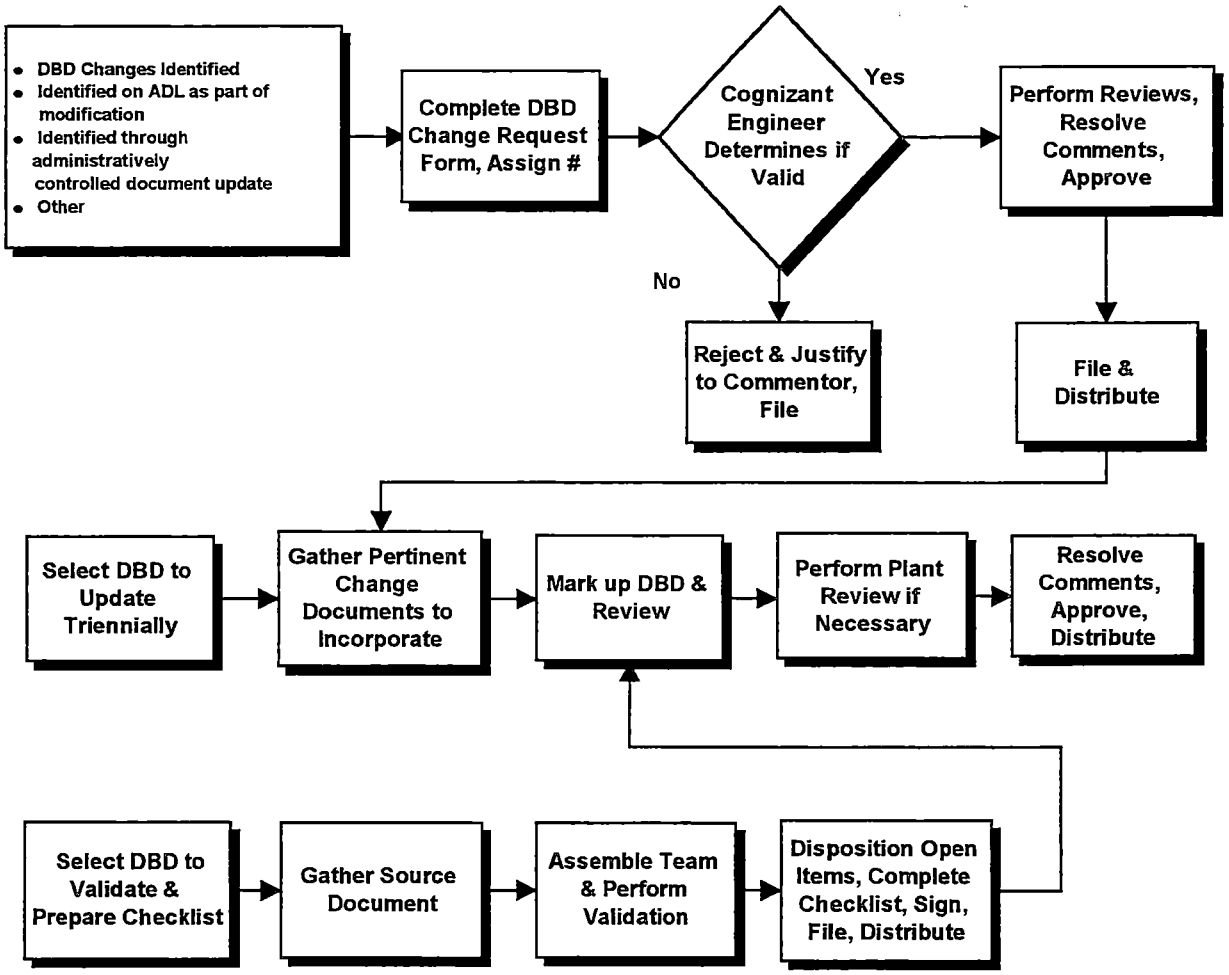
Onsite Review attributes include:

- Fulfillment of Technical Specifications requirements.
- Fulfillment of UFSAR requirements and commitments.
- Safety issues.
- Review of 10 CFR 50.59 Safety Evaluations for Technical Specification and UFSAR application.
- Procedural compliance.
- Administrative Radiological concerns.
- Fulfillment of station commitments to NRC, INPO, and other regulatory agencies.

CHECKS AND BALANCES

A 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.

Flowchart 10
Design Basis Document (DBD) Update Process
NEP-17-01



Design Basis Document (DBD) Update Process

NEP-17-01

PURPOSE

The DBD update process is used to evaluate DBD changes and incorporate approved changes. This process provides the controls to ensure that the change is appropriately reviewed, prior to updating the DBD.

PROCESS DESCRIPTION

A DBD change can result from a modification or it can be identified through the revision process associated with an administratively controlled document (such as an UFSAR change, setpoint change, etc.) or it can be self-initiated as part of the normal work process or as a result of a regulatory inspection or self-assessment.

Once an evaluation of a design change has determined that a DBD is affected, the DBD is indicated on the Affected Documents List (ADL) and the change is processed for an update evaluation. A DBD Change Request Form is initiated and placed into the review process. The process from this point on applies regardless of the reason for the originating change to the DBD. The review will determine if the change is valid for incorporation.

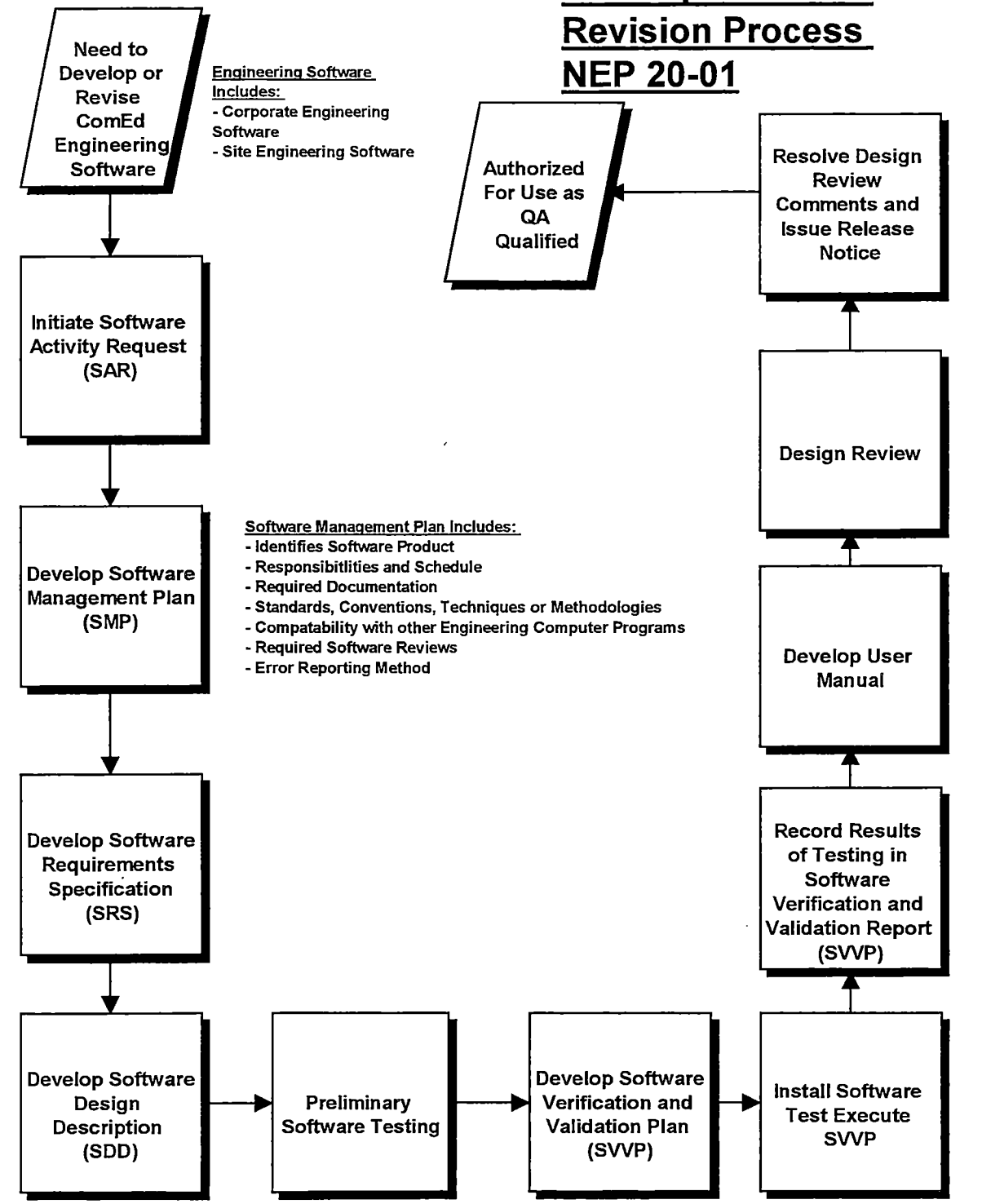
A final review and approval process will be completed and then it will be issued.

CHECKS AND BALANCES

The initial review by the Cognizant Engineer is used as the main determination for the validity of the change to the DBD. A triennial update will also be performed on selected DBDs to ensure that they are current.

An optional DBD Validation process which is performed by separate team is available for select DBDs. The results of this process are tied back to the update process described above.

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:

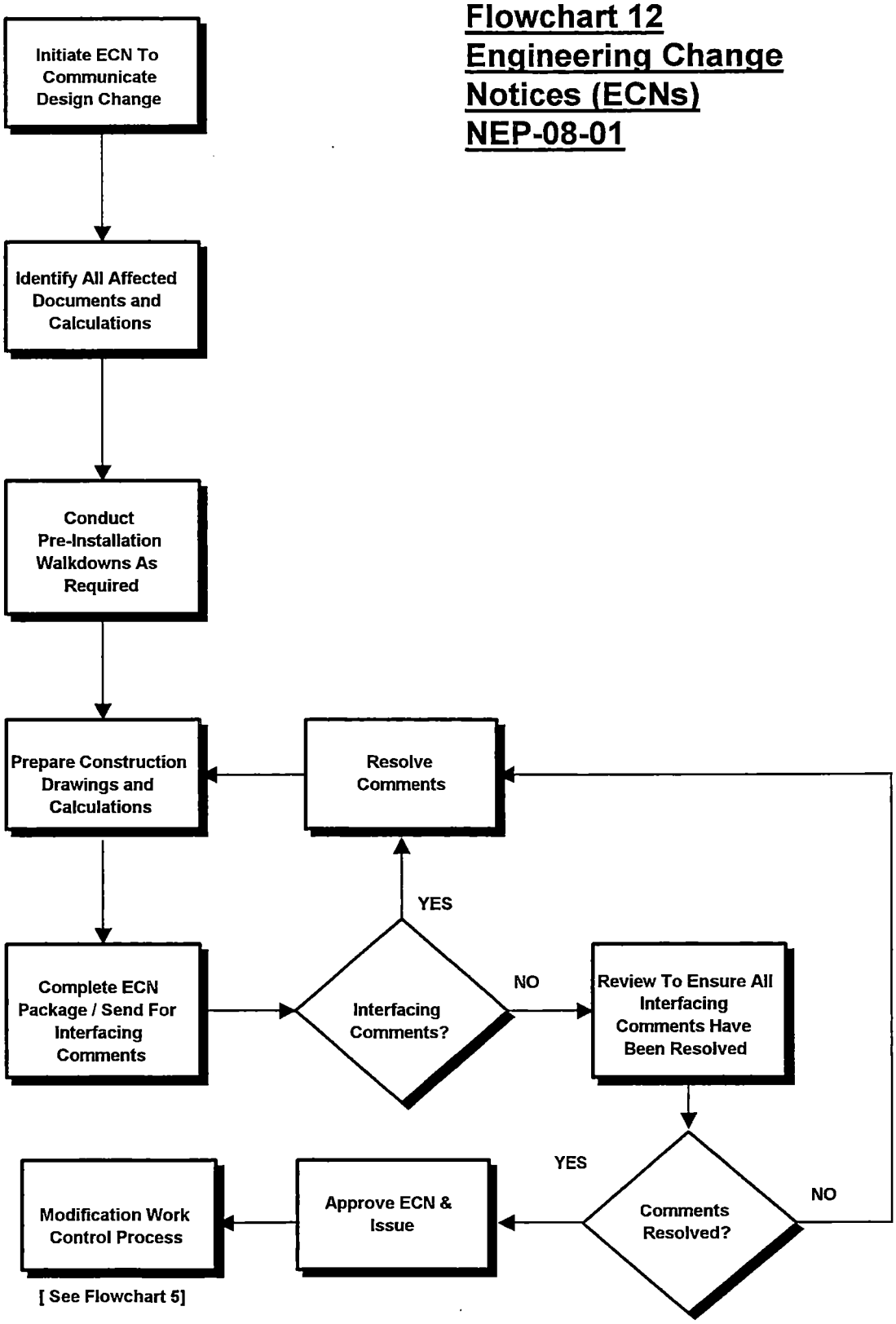
- tasks and criteria for accomplishing the Verification of the ECP.
- hardware and software configurations pertinent to V and V.
- traceable to both the software requirements and the software design.

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 Design Change Package. They are initiated through the Engineering 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

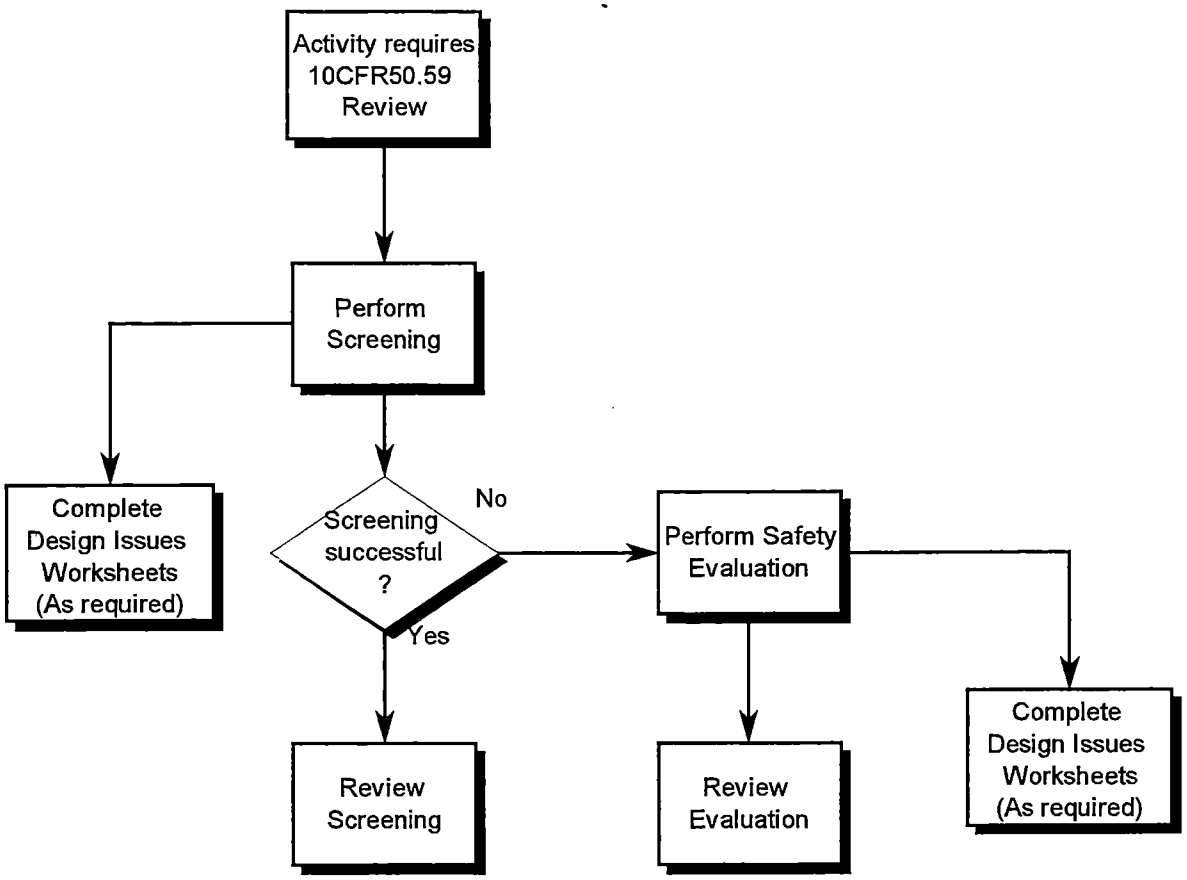
As the ADL is prepared through EWCS, all pending changes are identified and evaluated for their impact to the new change/addition. This allows for an additional evaluation of all previously planned changes and those which are currently underway.

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
DAP 10-02



Safety Evaluation Process

DAP 10-02

PURPOSE

Screenings and Safety Evaluations provide the basis for determining whether a Procedure Change, Test, Experiment, or Facility Change could; make changes to the facility or procedures as described in the UFSAR, conduct tests or experiments which are NOT described in the UFSAR, or involve a Unreviewed Safety Question or a change to the Technical Specifications.

Preparation

10 CFR 50.59 Screenings/Safety Evaluations are prepared by qualified individuals using detailed forms which address al provisions of 10 CFR 50.59.

Review

10 CFR 50.59 Screenings and Safety Evaluations are reviewed by individuals meeting the qualification requirements of ANSI N18.1-1971.

Additional Review of Safety Evaluations

Due to weaknesses in Dresden's performance in this area, all Safety Evaluations receive an additional review by a qualified individual designated by the Site Engineering Manager.. Additionally, all completed Safety Evaluations are independently reviewed by an Off Site Review Group comprised of knowledgeable individuals with various discipline backgrounds (Operations, Maintenance, Engineering, etc.). 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, the off-site Corporate Regulatory Assurance group, and the Engineering Oversight Team provide three levels of independent assessment of the quality and effectiveness of this process.

RECENT/PLANNED IMPROVEMENTS

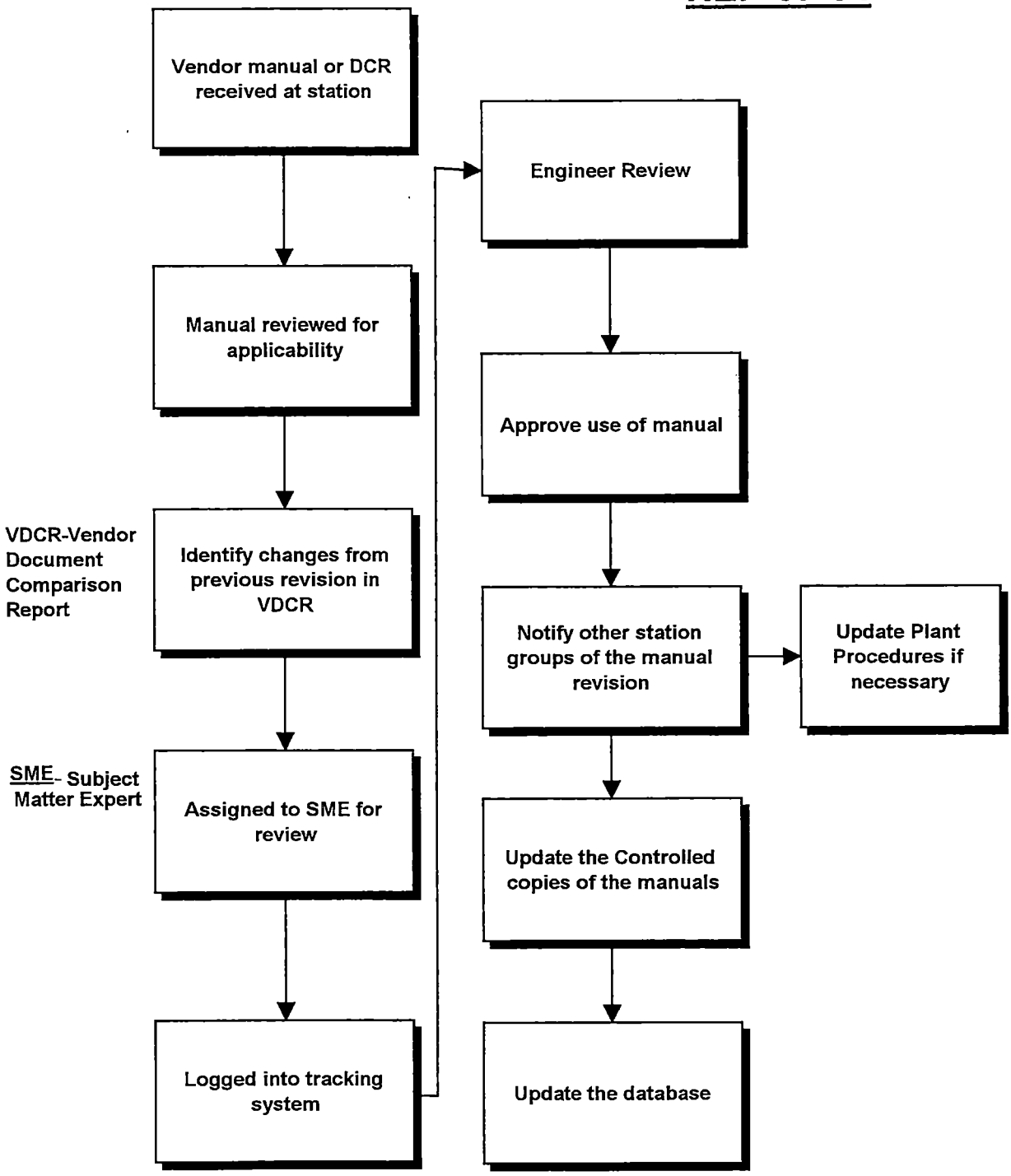
ComEd's Safety Evaluation Process has been the subject of several NRC Audits. The specific findings and ComEd's corrective actions are discussed in the station attachments. The following steps have been implemented on a corporate-wide basis to improve this process:

- 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.

- Corporate Regulatory Assurance performs an Offsite review of all Safety Evaluations

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

Flowchart 14
VETIP Processing
NEP-07-04



VETIP Processing

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 is received and processed through the VETIP Coordinator at the station. The following activities are performed for each vendor manual:

A review for applicability is 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.

For a technical change to an existing manual, a summary of revisions document, called a Vendor Document Comparison Report (VDCR), is prepared. The manual is routed to the Subject Matter Expert for approval.

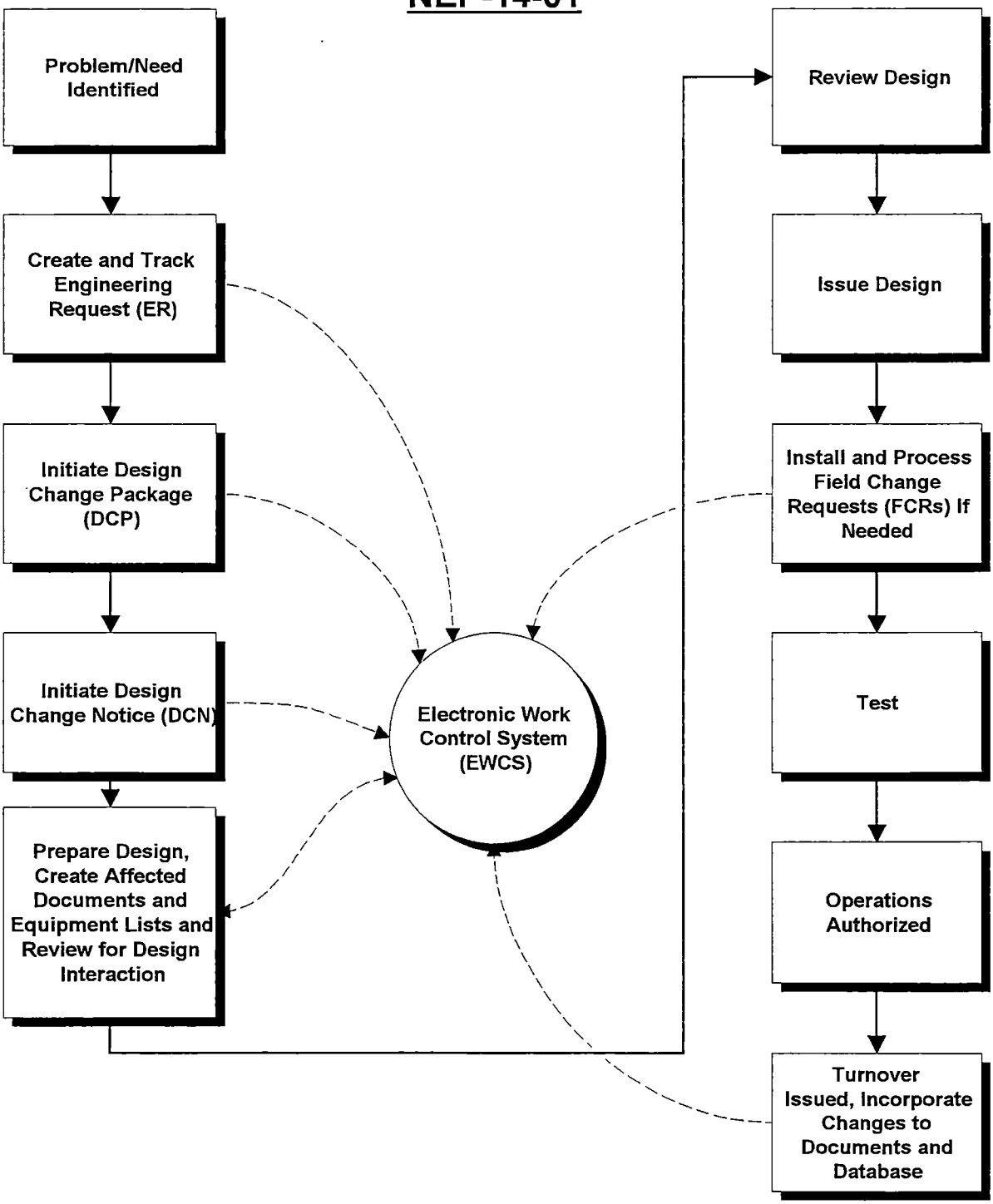
The SME determines what other station groups should be notified of the manual change and identifies which station procedures are affected. Following SME approval, the VETIP Coordinator processes the new vendor manual and updates hard copies and databases.

CHECKS AND BALANCES

The Subject Matter Expert ensures the right person is reviewing the manual.

Invoking a ComEd VETIP instruction ensure common processing at each station for better control and a more consistent review and documentation of VETIP information.

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
- Controlled Documents (CD)
- Equipment Database

These modules and their configuration control functions are outlined below.

PROCESS DESCRIPTION

Engineering Design Change Module

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

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

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

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

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

Document Change Requests (DCRs) - Used to document as found changes and discrepancies to design documents. DCRs use ADLs and AELs to identify and track the status of impacted documents and equipment data records through the change process. Note that a Turnover, not a

DCR, is the vehicle used to track closure of document and equipment data changes associated with DCPs and DCNs and is part of those respective processes.

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

Revision Tracking & Control (RT&C)

RT&C is the software portion of EDCM which controls equipment revisions and tracks changes.. RT&C provides the ability to change equipment data associated with an EDCM change object through an on-line process. Any authorized user 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 documents including drawings, calculations, procedures, and vendor information. The search features of CD are used by engineers and others to find and retrieve (from central files or through on-line viewing for some types of documents) these documents.

Equipment Database

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

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

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

CHECKS AND BALANCES

When a document is identified as affected by the change and is placed on the ADL, Engineering Design Change Module (EDCM) searches the document database for any other open change

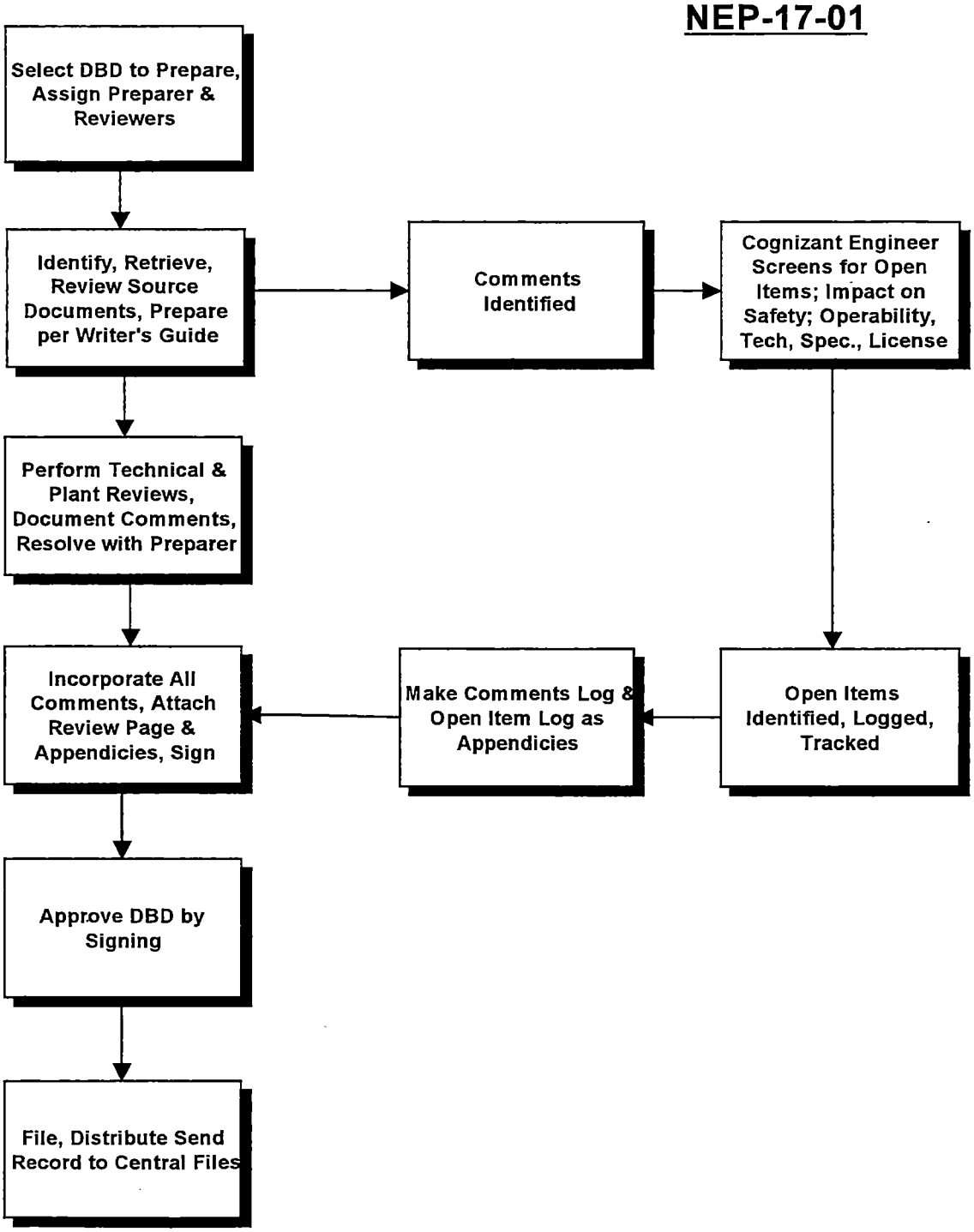
against the document and immediately notifies the user if found. This feature is also in place for equipment records placed on the AEL.

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.

Flowchart 16
DBD Development Process
NEP-17-01

DBD DEVELOPMENT



DBD Development Process

NEP-17-01

PURPOSE

The Design Basis Document, DBD, development process is controlled by a Writer's Guide, which provides guidance to the writers for consistent format and content. The process includes identifying original plant design basis, incorporating changes resulting from various types of modifications, reviewing existing design information, and resolving any conflicts between documents.

PROCESS DESCRIPTION

Engineers from the NSSS suppliers and Balance of Plant Architect Engineers, A/E's, were utilized in the development of the various DBDs. The NSSS writers access their internal sources to identify the references used to support the original design. The A/E writers access A/E project files and ComEd databases. In addition, they review all modifications to identify any impact on the design basis.

Reviews are performed by ComEd organizations and other A/E's that were involved in the design and operation of the Station. These groups include Site Engineering, System Engineering, Corporate Engineering, Nuclear Fuel Services, Mechanical & Structural Design, Electrical/Instrumentation & Control Design, and the Site Training departments. This provides a check to ensure the latest design information is identified.

When the review of a draft DBD is complete, comments are compiled and a meeting is held between the NSSS writers, A/E writers, the ComEd Engineers, and others that had significant technical input. Comments are discussed to identify discrepancies, assess their significance and determine a resolution. In some cases, where original studies or calculations are unavailable, system and component specifications as well as process flow diagrams are utilized to establish the original design basis. Where supporting calculations for modifications are incomplete, an open item is generated, evaluated for significance, and prioritized for resolution. References used in the DBDs to support the design basis are indexed and referenced in the DBD. When all comments have been addressed and the remaining open items logged and tracked, the DBD is issued.

In order to maintain the DBDs as living documents, a process is in place to ensure that any design changes are reviewed to determine their impact on the DBD. This process is addressed on Flowchart 10, DBD Update Process.

CHECKS AND BALANCES

Writers of DBD's are trained to recognize and report discrepancies during the writing process. DBD comments submitted by the writers and reviewers are screened by the cognizant ComEd engineer to determine their significance. Comments are either resolved and incorporated into the DBD or handled as discrepancies and prioritized for resolution. Evaluations to determine disposition of discrepancies were done by the Cognizant Engineer.

Discrepancies are evaluated and prioritized by the following Categories:

1. Safety Impact, Operability/Tech Spec Violation, Licensing Violation
2. Deficiency in Design Change
3. Resolution Required to Support Future Design Changes
4. Inconsistency or Missing Documentation that is not Necessary to Resolve

Category 1 items are immediately referred to the applicable ComEd process for performing Safety, Operability, and Reportability determinations.

Category 2 items are evaluated and short term action plans are developed for resolution.

Category 3 items are evaluated and long term action plans are developed to resolve the discrepancy.

Category 4 items are tracked via the Open Item Log, contained within the DBD, for resolution as part of on-going activities.

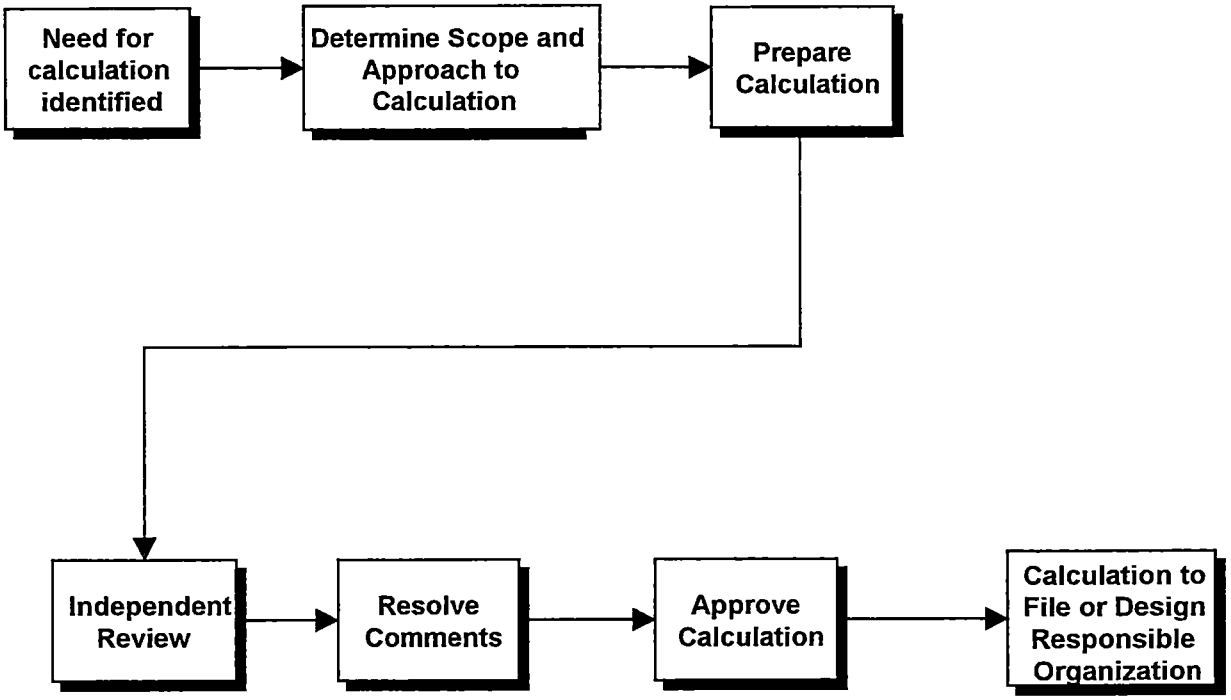
Cognizant DBD Engineers are responsible to track and resolve Open Items listed against their DBD. When appropriate actions are completed, the resolution is documented, any necessary DBD changes initiated, and the Open Item closed.

RECENT / PLANNED IMPROVEMENTS

Several process improvements have been made to the DBD program. Some are the result of assessment recommendations while others reflect lessons learned from experience. In May 1994 a detailed assessment effort resulted in the issuance of the Design Information Review Team (DIRT) Project Report. This report recommended the DBD Program coordinate more closely with the sites to better meet their needs, generate key missing data, and make licensing documentation more available to the users. As a result of these recommendations, several DBD content and format changes were incorporated to better meet the end users needs. In addition, the production of several topical DBD's was initiated.

In addition to these changes in the actual production process, there have been changes made in the administrative handling of DBD's and their associated references. A key change has been the indexing of DBD's and there references into the Electronic Work Control System, EWCS. EWCS is the computer database where all engineering documents are indexed and controlled. By utilizing this system, all engineers at the site can easily determine whether a DBD for a particular system or topic exists and can locate calculations referenced within the DBD. This improved accessibility should help foster increased use of DBDs during design related activities.

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.

Project files	Drawings
Meeting notes	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.

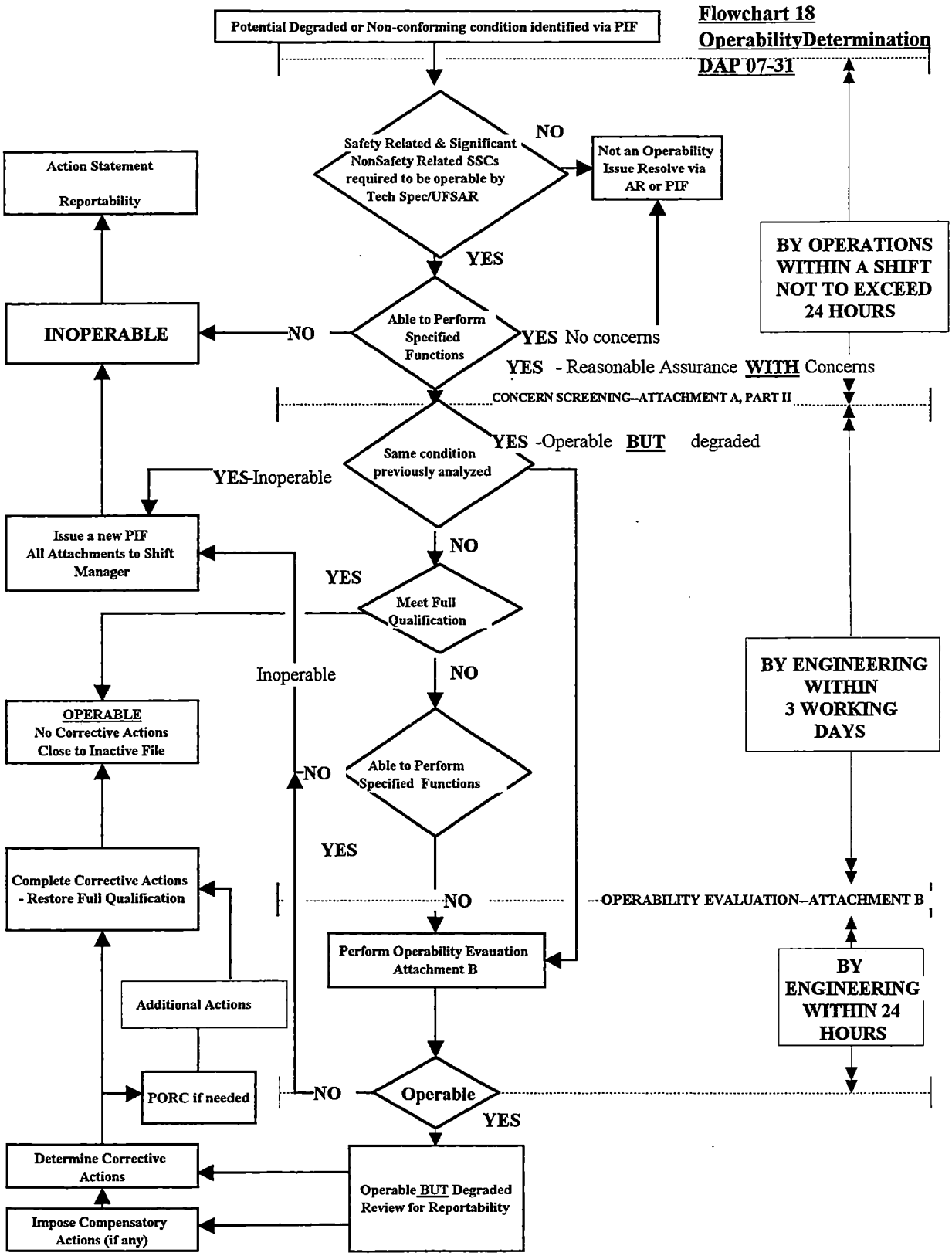
Qualification Testing Method

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.


Flowchart 18
OperabilityDetermination
DAP 07-31
Potential Degraded or Non-conforming condition identified via PIF
Safety Related & Significant NonSafety Related SSCs required to be operable by Tech Spec/UFSAR
NO
Not an Operability Issue Resolve via AR or PIF
YES
Able to Perform Specified Functions
NO
INOPERABLE
Action Statement
Reportability
YES No concerns
YES - Reasonable Assurance WITH Concerns
CONCERN SCREENING--ATTACHMENT A, PART II
Same condition previously analyzed
YES -Operable BUT degraded
YES-Inoperable
Issue a new PIF All Attachments to Shift Manager
NO
YES
Meet Full Qualification
Inoperable
NO
OPERABLE
No Corrective Actions
Close to Inactive File
NO
Able to Perform Specified Functions
YES
Complete Corrective Actions - Restore Full Qualification
NO
OPERABILITY EVALUATION--ATTACHMENT B
Perform Operability Evauation Attachment B
Additional Actions
PORC if needed
NO
Operable
YES
Determine Corrective Actions
Operable BUT Degraded Review for Reportability
Impose Compensatory Actions (if any)
BY OPERATIONS WITHIN A SHIFT NOT TO EXCEED 24 HOURS
BY ENGINEERING WITHIN 3 WORKING DAYS
BY ENGINEERING WITHIN 24 HOURS

Operability Determination Process

DAP 07-31

PURPOSE

Operability Determinations are performed when the capability of a system, structure, or component (SSC) to perform its specified function(s) as required by the Technical Specifications or UFSAR cannot be unequivocally demonstrated, or where a degraded or nonconforming condition results in a judgment that the equipment is operable but that there are remaining concerns or uncertainties. This is applicable to Safety Related SSCs and significant Non-Safety Related SSCs which are relied upon to remain functional during and following design bases events.

Issue Screening

When an operability issue is identified, the Shift Manager/Unit Supervisor immediately performs a detailed Issue Screening. The attributes of this screening are as follows:

1. Does the affected SSC receive/initiate an RPS or ESF actuation signal?
2. Is the affected SSC in the main flow path of an ECCS or support system?
3. Is the affected SSC used to:
    - Maintain containment integrity?
    - Shutdown the Reactor?
    - Maintain the Reactor in a shutdown condition?
    - Prevent or mitigate the consequences of an accident that could result in off-site exposures comparable to 10 CFR 100 guidelines?
4. Does the SSC provide required support (i.e. cooling, lubrication, etc.) to a TS required SSC?
5. Is the SSC used to provide isolation between safety trains, or between safety and non-safety interfaces?
6. Is the SSC required to be operated manually to mitigate a design basis event?
7. Does the issue place the SSC outside current or pending UFSAR design requirements?
8. Is there reasonable assurance that the SSC is capable of reliably performing it's SPECIFIED FUNCTION in this condition? (Review LCO, Degraded Equipment Log, Temporary Alterations Log, Abnormality Log, Open Operability Determinations, etc.)

Completion of the Issue Screening will determine whether the SSC is Operable with no concerns; Inoperable, in which case it is reviewed for reportability; or Operable with potential concerns. A determination of operability with potential concerns will require a Concern Screening to be performed by Engineering. Issue Screenings are performed as soon as possible, usually within one 8 hour shift, up to 24 hours after PIF initiation.

Concern Screening

Concern Screenings are performed by knowledgeable qualified Engineers to determine their is an Operability Concern. Screenings are performed using detailed guidance. Guidance attributes include:

1. Does the issue place the SSC outside applicable codes or standards or other Licensing Basis requirements or other NRC requirements?
2. Does the issue involve operating experience information or Engineering reviews that demonstrates a design inadequacy?

3. Does the issue involve documentation required by the NRC such as 10 CFR 50.49 (EQ), Fire Protection, ATWS, SBO, etc. being NOT available or inadequate?
4. Is there reasonable assurance that the SSC is capable of reliably performing it's specified function in this condition?

Completion of the Concern Screening will determine whether the SSC is Operable or Inoperable, or whether the concern is confirmed. If the latter, an Operability Evaluation is performed. Concern Screenings are completed within 3 days after the Operations screening has determined the SSC to be "Operable with Potential Concerns."

Operability Evaluation

Operability Evaluations are performed by knowledgeable qualified Engineers using detailed guidance. Guidance attributes include:

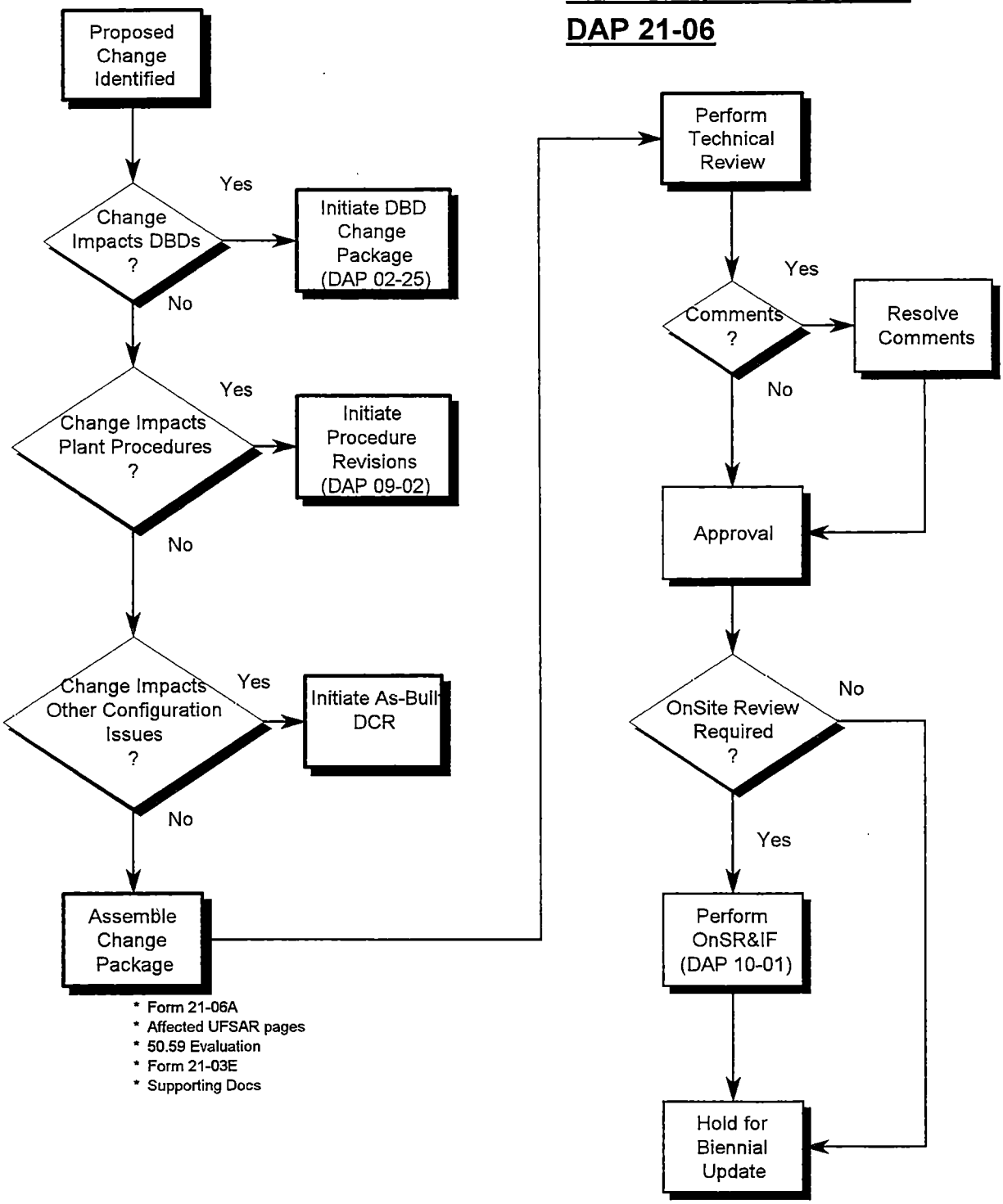
1. Does the affected SSC receive/initiate an RPS or ESF actuation signal?
2. Is the affected SSC in the main flow path of an ECCS or support system?
3. Is the affected SSC used to maintain containment integrity, shutdown the reactor, maintain the reactor in a shutdown condition, or prevent or mitigate the consequences of an accident that could result in off-site exposures comparable to 10 CFR 100 guidelines?
4. Does the SSC provide required support (i.e. cooling, lubrication, etc.) to a TS required SSC?
5. Is the SSC used to provide isolation between safety trains, or between safety and non-safety interfaces?
6. Is the SSC required to be operated manually to mitigate a design basis event?
7. Impact on the Technical Specifications.
8. Impact on the UFSAR.
9. Impact on other Licensing documents.
10. Is the SSC degraded or non-conforming?
11. Can the SSC meet its specified functions?

Completion of the Operability Evaluation will determine if Compensatory Actions are required to maintain functionality; or if corrective actions are required to restore full qualification. The Operability Evaluation is reviewed by the Engineering Superintendent/Designee, a 10 CFR 50.59/Operability Review Group Member, the Regulatory Assurance Supervisor and the Shift Manager/Unit Supervisor. The Shift Operations Supervisor (SOS) then decides if a follow-up review by the Plant Operation Review Committee (PORC) is required.

If PORC review is required, it will normally occur during the next scheduled PORC unless otherwise required by the SOS based on the significance of the evaluation. PORC review is a safety/quality review of the Engineering evaluation and/or determination of operability by the Shift Manager/Unit Supervisor and is not part of the in-line approval. The PORC will review the proposed corrective actions, determine the timeliness for the corrective actions and the need for a 10 CFR 50.59 review and/or UFSAR revision, and the scope and timeliness for compensatory actions.

An Operability Determination is open as long as the degraded or non-conforming condition exists which deviates from the design or licensing basis while the SSC remains operable per the evaluation. The operability can only be closed when it can be shown that the SSC has been repaired or modified to meet the original full qualification or the design basis has been changed via Plant Design Change and/or UFSAR change so that the as-found condition now meets Full qualification. Operability Evaluations are completed within 24 hours after it has been "Concern" confirmed..

Flowchart 19
UFSAR Update Process
DAP 21-06



UFSAR Update Process

DAP 21-06

PURPOSE

Changes made to the facility, equipment, analysis, procedures, programs, or organizations which change the description included in the UFSAR, require a UFSAR Change to be initiated. UFSAR Changes are controlled through detailed preparation and review processes as described below.

Change Preparation

The initiator of a UFSAR Change thoroughly researches the change to ensure that it does not:

- Impact the Technical Specifications.
- Impact the Design Basis Documents.
- Impact Plant procedures.
- Impact system design.
- Impact Station commitments.
- Impact other Configuration Management Issues/Programs.

10 CFR 50.59 Safety Evaluation

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

- All UFSAR Changes receive a 10 CFR 50.59 Safety Evaluation.
- 10 CFR 50.59 Screenings and Safety Evaluations are performed and reviewed by qualified individuals.

Technical Review

UFSAR Changes which are determined to be "Technical Changes" receive a Technical Review to verify that the proposed information is technically correct. Technical Reviews are performed by individuals knowledgeable in the subject matter

Technical Changes include:

- Procedure changes.
- Changes to Controlled programs, e.g., ISI/IST, EQ.
- NRC Correspondence, e.g., SERs, Bulletins, Generic Letters.
- Effects of tests or experiments not currently described in the UFSAR.
- Technical Specification changes.

Onsite Review & Investigative Function (OnSR&IF)

UFSAR Changes which may involve an Unreviewed Safety Question or affect nuclear safety receive a critical and thorough administrative Onsite Review. Onsite Reviews are performed by at least two individuals who collectively possess background and qualification in the subject matter.

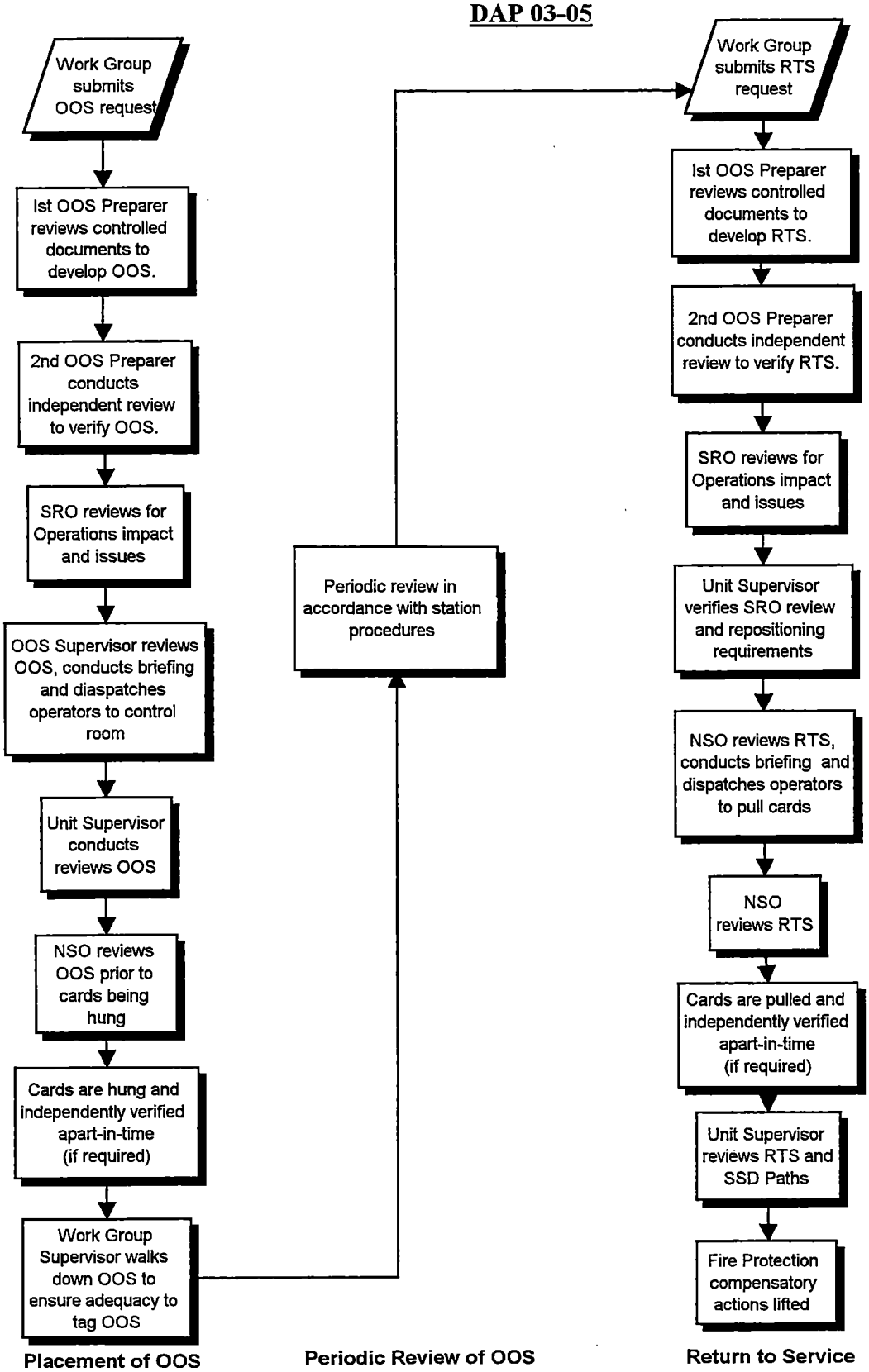
The Onsite Review encompasses:

- Fulfillment of Technical Specifications requirements.
- Fulfillment of UFSAR requirements and commitments.
- Safety issues.
- Review of 10 CFR 50.59 Safety Evaluations for Technical Specification and UFSAR application.
- Procedural compliance.
- Administrative Radiological concerns.
- Fulfillment of station commitments to NRC, INPO, and other regulatory agencies.

Approval

UFSAR Changes are reviewed and Approved by the Cognizant Engineering Supervisor.

Flowchart 20
Out of Service/Return to Service
DAP 03-05



Out Of Service/Return To Service Process

DAP 03-05

PURPOSE

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

PROCESS DESCRIPTION

The following is an outline of the equipment Out-Of-Service (OOS) and Return to Service (RTS) process.

PLACEMENT OF OOS

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 ComEd's Electronic Work Control System (EWCS).

1. Work Groups requesting the OOS are responsible to sufficiently define the scope of the work to allow the Operations Department to develop an adequate OOS.
2. Qualification requirements are established for individuals who prepare and review OOS. Controlled documents and drawings are used to ensure accuracy of prepared OOS. When controlled drawings are unavailable, the OOS will be walked down in the field to ensure accuracy. A second qualified OOS Preparer independently verifies the OOS as correct.
3. The OOS is reviewed by an SRO licensed operator to identify Technical Specification (Tech Spec), Primary/Secondary Containment related, fire protection/Appendix R and other issues.
4. OOS Supervisor conducts briefing of OOS, then sends operators into control room.
5. A SRO licensed Unit Supervisor in the Control Room conducts an independent review and weighs the impact of the OOS on the probabilistic risk assessment for the Unit.
6. 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.
7. Both licensed and non-licensed operators may place OOS cards. Cards are hung and then independently verified.
8. The Work Group Supervisor is responsible to verify the OOS has been correctly hung and is adequate for the scope of the work.

PERIODIC REVIEW OF OOS

At Dresden, a quarterly review of OOS cards is performed for safety related SSCs. At least 3%, if available, of the OOS cards are verified. Moreover, all safety-related OOSs that are not in a high radiation area and are over 90 days old are also reviewed. For Non-Safety Related SSCs, the quarterly review covers three randomly selected OSSs which have been hung in the last 3 months and are still in place. In addition to the quarterly reviews, OOSs greater than one year old are reviewed annually to determine if the OOS is still necessary and are walked down to verify that cards are still in place..

RETURN TO SERVICE

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

1. A qualified OOS Preparer reviews controlled documents and drawings to prepare the RTS and determine repositioning requirements for equipment.
2. A second OOS Preparer verifies the RTS is correct.
3. RTS is SRO reviewed to identify potential Tech Spec/Containment issues.
4. RTS is verified by the Unit Supervisor to ensure Tech Spec/Containment issues have been identified and that equipment repositioning requirements are appropriate.
5. A NSO will review the RTS and brief the operators who will reposition equipment and remove the OOS cards.
6. Equipment is repositioned and OOS cards are removed and independently verification.
7. The Unit Supervisor reviews the RTS to restore Safe Shutdown Paths and to ensure all actions are properly completed.

CHECKS AND BALANCES

Independent verification is used throughout the OOS program. There are 2 OOS Preparers and each is responsible to independently review controlled documents and drawings to satisfy themselves that the points of isolation and special instructions are correct. Technical Specification, 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 RTS, 2 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.

RECENT/PLANNED IMPROVEMENTS

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

Appendix III - Nuclear Fuel Services' Design Processes

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

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

Organization and Responsibilities:

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

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

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

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

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

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

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

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

Core Reload Design 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)

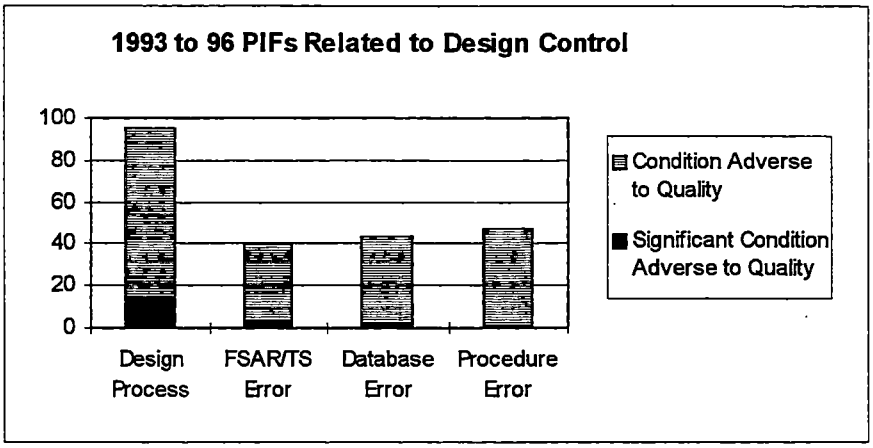
A review was performed of NFS generated PIFs from 1993 (the first year the PIF process was used in NFS) to present (November 8, 1996). As described in Action (d), the PIF process is common to all six nuclear stations and is also used by NFS to identify, document, assess, and correct design bases and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

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

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

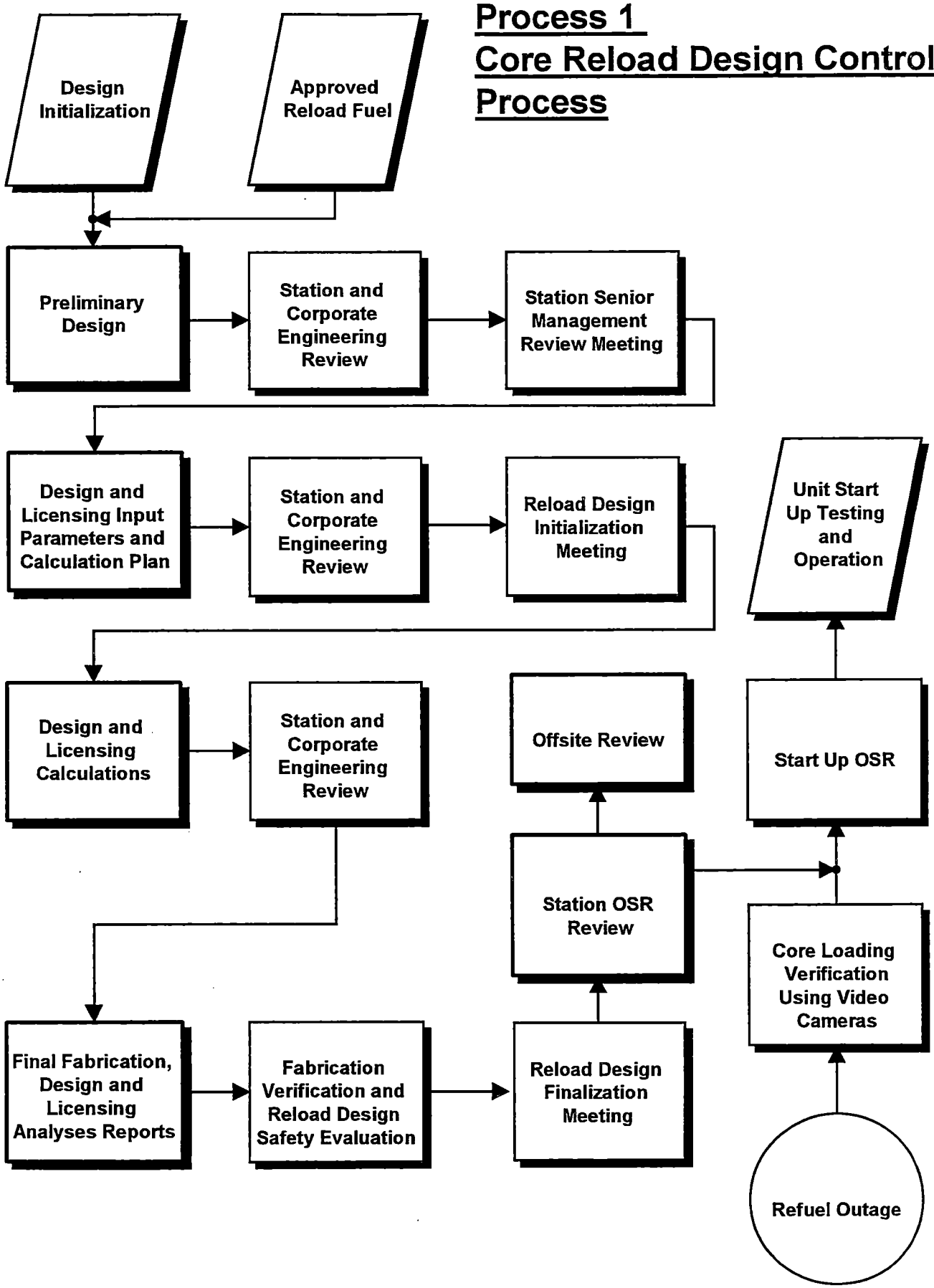
However, weaknesses were also identified such as:

> "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."

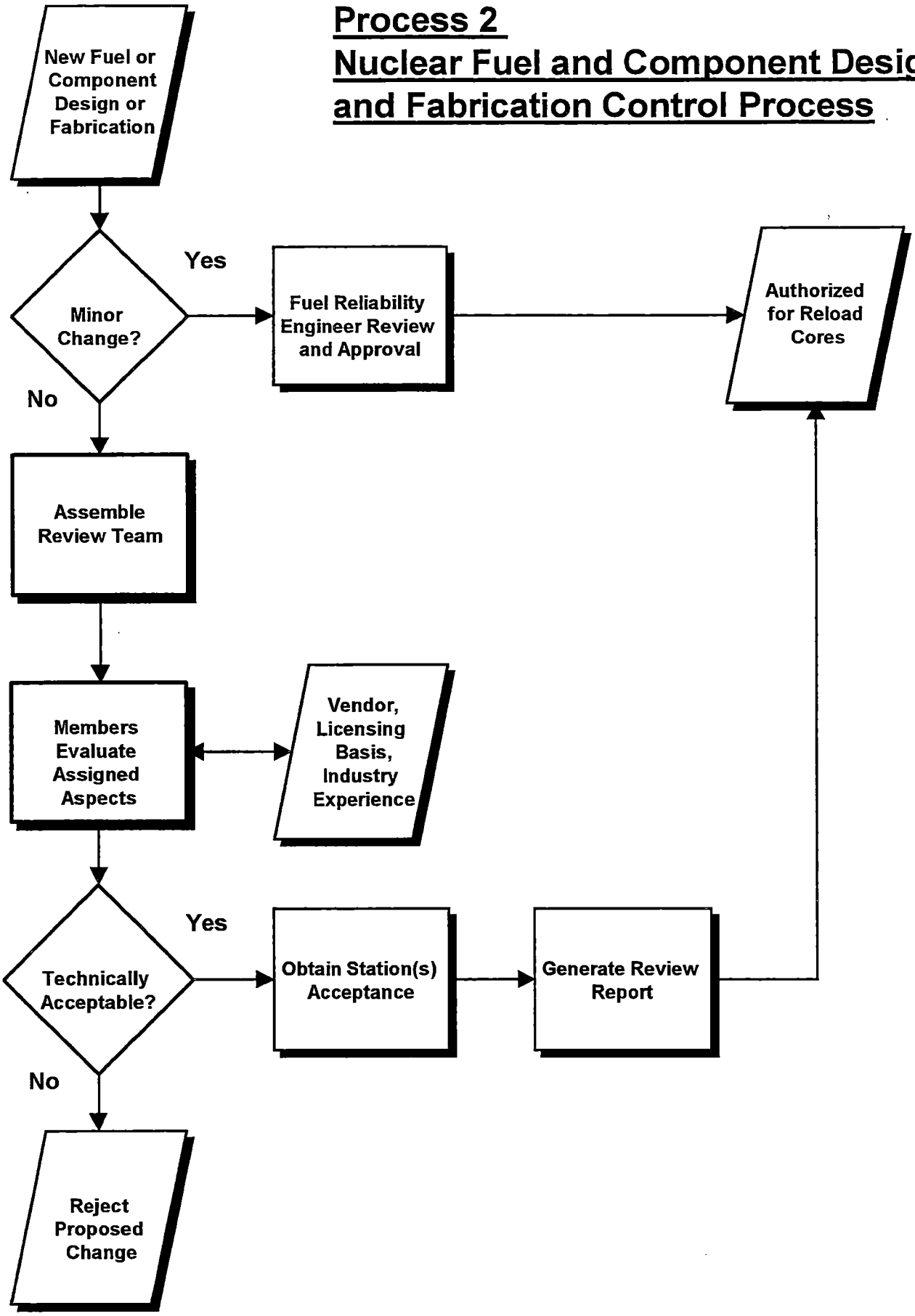
These weaknesses have been and are continuing to be addressed through enhancements to the reload design process.

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