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Docket Number 50-346

License Number NPF-3

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Document Control Desk
United States Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Subject: Response to NRC Request for Information Regarding the Adequacy and Availability of Design Bases Information

Ladies and Gentlemen:

This letter provides Toledo Edison's (TE) response for the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS), to the Nuclear Regulatory Commission's (NRC) "Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Adequacy and Availability of Design Bases Information", (Log Number 4298) dated October 9, 1996. The request for information requires the following information be provided to the NRC by February 12, 1997:

(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;

(b) Rationale for concluding that design bases requirements are translated into operating, maintenance, and testing procedures;

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

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

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

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Toledo Edison believes that the programs and assessments, as described in the attachment, provide reasonable assurance that: system design basis information and system configuration have been captured and translated into plant procedures; programs are in place to evaluate and control changes; the systems perform and are tested appropriately; and identified conditions adverse to quality are evaluated and appropriately corrected.

The DBNPS response to Items (a) through (e) are contained in the attachment. The information provided is intended to describe currently existing processes and procedures. It is not intended to preclude subsequent changes following normal practices, or to require NRC notifications or approvals of such changes other than those currently required. This response does not modify any prior NRC commitments, but does create a commitment to complete the DBNPS Design Basis Validation Program as described below.

The NRC's letter further requests whether or not the licensee has undertaken any design review or reconstitution programs, and if not, to explain the rationale for the decision.

The DBNPS has completed much of the effort required to compile the DBNPS design basis. This effort, as described in the attachment, meets the intent of NUMARC 90-12, "Design Basis Program Guidelines". This has been accomplished through the establishment of the DBNPS "Configuration Management Program" and through the development of "System Descriptions" and the "Design Criteria Manual".

It is recognized that the documentation of existing DBNPS design calculations and supporting assumptions, although consistent with industry standards at the time of original design and construction during the late 1960s through the mid 1970s, may not contain the level of detail of today's standards. In a continuing effort to strengthen the design basis, the DBNPS has commenced a Design Basis Validation Program. The purpose of this program is to provide further assurance that design basis information, specifically design basis calculations, are consistently reflected in the physical plant and those controlled documents used to support plant operation, and contain sufficient information to support the underlying assumptions contained in the calculations.

The validation effort will be conducted consistent with the guidance of NUMARC 90-12. The program will cover, at a minimum, the Maintenance Rule risk significant systems and selected topical areas, such as: seismic; missile; flooding; and high energy line breaks. The plan and time frame for accomplishing this activity will be transmitted to the NRC under separate letter by March 31, 1997.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs at (419) 321-8466.

Very truly yours,

JCS/RJE:laj

Enclosure
Attachment

cc: A. B. Beach, Regional Administrator, NRC Region III
S. J. Collins, Director, Office of Nuclear Reactor Regulation
A. G. Hansen, DB-1 NRC/NRR Project Manager
S. Stasek, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

RESPONSE

TO

10 CFR 50.54(f) REQUEST FOR INFORMATION

FOR THE

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

This letter is submitted pursuant to 10 CFR 50.54(f). Attached is Toledo Edison's response to the "Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Adequacy and Availability of Design Bases Information" for the Davis-Besse Nuclear Power Station, Unit Number 1.

By: ______________________________
J. P. Stetz, Senior Vice President - Nuclear

Sworn and Subscribed before me this 11th DAY OF FEBRUARY, 1997

____
Notary Public, State of Ohio

JOSEPH C. SZWEJKOWSKI
Notary Public, State of Ohio, Cuy. Cty.
My Commission Expires July 21, 2001

Docket Number 50-346
License Number NPF-3
Serial Number 2438
Attachment

RESPONSE TO THE REQUEST

FOR INFORMATION REGARDING

ADEQUACY AND AVAILABILITY

OF

DESIGN BASES INFORMATION

RESPONSE TO THE
REQUEST FOR INFORMATION REGARDING
ADEQUACY AND AVAILABILITY OF DESIGN BASES INFORMATION
FOR
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NUMBER 1

NRC Request for Information:

(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.

Toledo Edison Response:

A.1.0 10 CFR 50 Appendix B, Nuclear Quality Assurance Requirements

The Davis-Besse Nuclear Power Station's (DBNPS) "Nuclear Quality Assurance Program" complies with the criteria established by 10 CFR 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", as described in the DBNPS "Updated Safety Analysis Report" (USAR) Section 17.2, "Quality Assurance During the Operations Phase". The DBNPS site organization is responsible for the implementation of the "Nuclear Quality Assurance Program". To assist in implementation of the "Nuclear Quality Assurance Program", the "Nuclear Quality Assurance Manual" (NQAM) was developed in the form of a functional manual that implements the requirements of 10 CFR 50 Appendix B.

The DBNPS Nuclear Group (NG) procedures were developed as interface procedures for those activities that are common to more than one department. These procedures reflect the "Nuclear Quality Assurance Program" description as described in USAR 17.2, the NQAM, and applicable regulatory requirements and prescribed requirements associated with Nuclear Group interfacing activities. Departmental procedures are organizationally unique documents that implement the requirements of the NQAM and applicable Nuclear Group procedures for the departments. Department procedures and revisions are reviewed and approved by the responsible department director. Procedures are also approved or concurred with by the Manager - Quality Assessment on a selective basis for those activities within the scope of the "Nuclear Quality Assurance Program".

The DBNPS engineering design and configuration control processes manage and coordinate activities that evaluate, change, and update the DBNPS documents and data bases. These documents and data bases contain the information describing the as-built critical characteristics of structures, systems, and components at the DBNPS. The requirements and responsibilities for maintaining engineering design and configuration control are contained in DBNPS procedures NG-EN-00307, "Configuration Management", and NG-EN-00301, "Plant Modifications". These procedures use implementing procedures to

maintain configuration control documents and configuration data bases, design bases, regulatory requirements, and the as-built physical plant configuration. The procedures also include requirements related to modifying or abandoning plant equipment. The upper tier and implementing procedures were developed, in part, to comply with 10 CFR 50.59, "Changes, Tests, and Experiments"; 10 CFR 50 Appendix B Criterion III, "Design Control"; ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"; and ANSI/ANS 3.2-1982, "Administrative Controls and Quality Assurance for Nuclear Power Plants". Commitments to industry standards are described in USAR Table 17.2-1, "Applicable NRC Regulatory Guides, ANSI Standards, and Industry Codes".

A.2.0 Design Control Process

The purpose of the DBNPS engineering design control process is to provide assurance that changes to plant design are accurately performed, adequately reviewed, and that the applicable information is captured and translated into the appropriate design and configuration control documentation. The purpose of design control measures are to provide assurance that the design activities are carried out in a planned, controlled, orderly manner and that design and regulatory requirements, as well as appropriate quality standards, are correctly translated into design, procurement, and procedural documents. The design control process also provides design control measures that are commensurate with those applied to the original design and are approved by the organization that performed the original design or another qualified organization, or by qualified personnel at the DBNPS. Design activities at the DBNPS are carried out in accordance with the requirements of ANSI N45.2.11 and ANSI/ANS 3.2.

The design control measures involve: 1) preparation of a conceptual design package if required; 2) performance of a safety review/evaluation; 3) providing the design input; 4) performance of the design process; 5) performance of the design analysis and design verification; 6) controlling the design interface and design reviews; and 7) controlling changes to design. The design control measures also include activities such as: field design engineering; physics, seismic, stress, thermal, hydraulic, radiation, and USAR Chapter 15 accident analysis; compatibility of materials; accessibility for inservice inspection, maintenance and repair; and delineation of acceptance criteria for tests and inspections.

A.2.1 Design Modifications

The modification (MOD) process, controlled via procedure NG-EN-00301, "Plant Modifications", provides the methodology which engineering employs to develop plant modifications. The MOD process incorporates elements of the document EPRI TR-103586, "Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants". The MOD process includes steps in the detailed design review which identify any impact on the design bases of affected systems. This review is documented in the Design Report for Normal MODs, or the Design Summary for Limited MODs (minor in scope and complexity). As part of this detailed design review, a design basis review is performed including calculations. If an assumption contained in a calculation cannot be confirmed, or agreed to as a conservative and

reasonable approach, an alternate design approach is pursued to establish the basis for the final design. A safety review/evaluation is completed prior to implementation of the MOD.

Recognition and evaluation of a MOD's potential impact on various design-related programs (for example, Radiation Protection, Fire Protection, Seismic Qualification, Environmental Qualification, etc.,) is performed via the use of a Program Impact Review Checklist. The identification and update of other potentially affected programs or configuration documents is also supported through an Interdepartmental Review (IDR) process.

Modifications affecting structures, systems, or components (SSCs) with functions important to safe operation also receive design verification in compliance with ANSI N45.2.11. The design verification function is performed to provide assurance that the MOD design meets the specified design inputs. Design documents that require verification are identified in order that the verification process is completed prior to release for procurement, manufacturing, modification, or to other organizations for use in other design activities. Any changes to previously-verified designs are also verified. Design verifications and methods used are documented, including the identification of the individuals performing the verification.

As part of the IDR process, the completed MOD design package is reviewed by departments affected by the design prior to the release of the package. Review comments are documented, resolved, and analyzed for potential impact on safety evaluations and design verifications.

A.2.2 Field Problem Resolutions

The modification process also provides a mechanism, Field Problem Resolutions (FPRs), for site personnel to communicate problems to engineering that are associated with either implementing a modification or performing maintenance via a Maintenance Work Order (MWO). An FPR is a document used to identify documentation or field problems related to in-process maintenance activities (Replacement or Design Change FPRs), modification activities (MOD FPR), or setpoint change requests (Setpoint Change FPRs), and obtain an engineering resolution to the problem. Field Problem Resolutions require completion of a safety review/evaluation prior to returning the system to service.

The FPR identifies potentially affected documents and provides for an interim and/or final resolution to the problem. It also includes identification of the action taken by engineering and the documents associated with the resolution, and is used for implementation of the resolutions when field work is required. These FPR processes are governed by the following procedures:

1) Modification FPRs NG-EN-00301, "Plant Modifications"

2) Setpoint Change FPRs NG-EN-00310, "Setpoint Control"

3) Replacement FPRs NG-DB-00205, "Plant Maintenance"

4) Design Change FPRs NG-DB-00205, "Plant Maintenance"

A.2.3 Temporary Modifications

Changes to plant design that are not to be permanently installed are controlled via procedure NG-EN-00313, "Control of Temporary Modifications" (TM). A TM is a temporary alteration to in-service plant equipment or systems that do not conform with approved drawings or other design documents (for example, electrical jumpers, disabled annunciator alarms, installed/removed blank flanges). Oversight is provided for the purpose of ensuring that these modifications are temporary and are not considered a permanent modification. For example, Plant Engineering is required to perform an evaluation and obtain Plant Manager concurrence for a TM to remain in place greater than 180 days.

The TM process requires that changes to affected procedures reflecting the TM are completed prior to returning the system to service. Drawings, except short duration TMs of less than 14 days, are also required to reflect the TM prior to returning the system to service. A 10 CFR 50.59 safety review/evaluation and appropriate training of the TM are completed prior to returning the system to service.

A.2.4 Implementation of Design Changes

Physical alterations to DBNPS structures, systems, and components (SSC) are implemented by the Maintenance Work Order (MWO) process. The MWO process is governed by procedure DB-DP-00007, "Control of Work". This procedure provides guidance on the field implementation of a design change. It also provides guidance on when a design change (for example, Design Change FPR) is required during the normal course of a maintenance activity.

A.3.0 Configuration Control Process

The purpose of the configuration control process at the DBNPS is an integrated management process to ensure that the plant's physical and functional characteristics are maintained in conformance with the plant's design bases. These requirements apply to SSCs, including office building equipment, which are:

1) Safety-related; or

2) Part of the Augmented Quality (AQ) Program; or

3) Fire Protection components which may adversely affect any portion of the Plant Fire Protection System; or

4) Any other SSCs which may affect safe operation, safe shutdown, or are otherwise required to support power production.

A listing of documents and data bases that contain configuration control information is provided in procedure NG-EN-00307, "Configuration Management", Attachment 1, "Configuration References".

A sample of documents and data bases contained in this attachment are as follows:

1) Procedures

2) System Descriptions

3) Quality Classification List

4) Toledo Edison Regulatory Management System (TERMS)

5) Vendor Drawings

6) Vendor Manuals

7) Piping & Instrument Diagrams (P&IDs)

8) Operational Schematics (OS)

9) Updated Safety Analysis Report (USAR)

10) Davis-Besse Configuration Equipment Summary (DBCES) data base

11) Design Criteria Manual

When a change is made to any of these configuration control documents or data bases due to a change to the plant, other configuration control documents are reviewed to provide assurance that the change is consistently represented. Procedure NG-EN-00307, "Configuration Management", Attachment 1, "Configuration References", provides assistance to site personnel in determining documents or data bases that may need to be changed. This review provides for the changed information to be is consistently represented on configuration control documents and data bases. Configuration control documents and data bases affected by modification activities associated with procedure NG-EN-00301, "Plant Modifications", are required to be updated within the time period assigned in procedure NG-EN-00307, "Configuration Management". For example, changes to procedures, design drawings, and the USAR must be effective prior to returning the system to service.

A.4.0 Control of Procedures

Activities affecting quality are prescribed by procedures and instructions of a type appropriate to the circumstances, and are accomplished in accordance with these documents. These documents include the necessary operating limits and tolerances on materials, equipment, processes, and procedures for activities from design through operation.

Methods and requirements for the preparation, processing, and control of Nuclear Group, Department and Section/Unit procedures are governed by procedure NG-NA-00115, "Control of Procedures". This procedure was developed using the guidance from:

1) 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures and Drawings;"

2) Regulatory Guide 1.33-1972, "Quality Assurance Program Requirements (Operations)", Appendix A; and

3) ANSI/ANS 3.2-1982, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

Procedure NG-NA-00115, "Control of Procedures", applies to Nuclear Group and external support personnel involved in the preparation, processing, and control of DBNPS procedures. Changes to procedures are also integrated with the configuration management process by requiring procedure changes to be made concurrent with other configuration control documents and data bases in accordance with procedure NG-EN-00307, Configuration Management". Changes to procedures require a 10 CFR 50.59 safety review/evaluation and a review/evaluation for changes to regulatory commitments. The procedures describing these reviews are NG-EN-00304, "Safety Review and Evaluation" and NG-NS-00802, "Commitment Management."

The "Procedure Writers Guidelines" (PWG) supplements the instructions provided in procedure NG-NA-00115, "Control of Procedures", for preparation of Nuclear Group, Department, and Section/Unit procedures. Standard information on procedure format, style, and step development is provided to achieve clarity and uniformity of procedures for the DBNPS and external support groups.

The "Procedure Validator's Guidelines" (PVG) are provided to assist the sponsoring organization in fulfilling the procedure validation requirements of procedure NG-NA-00115, "Control of Procedures". Where applicable, the PVG provides guidance for personnel involved in the procedure validation process to achieve consistent quality of validation.

Guidance is included for:

1) Selection of the appropriate validation method

2) Selection of validation team members

3) Actual performance of procedure validations

4) Use of a Procedure Validation Checklist

A.5.0 10 CFR 50.59 Changes, Tests and Experiments

The process that implements 10 CFR 50.59 at the DBNPS is described in procedure NG-EN-00304, "Safety Review and Evaluation". This process is a two-tier process; the first tier consists of a safety review and the second tier consists of a safety evaluation. A safety review is a screening review of a proposed permanent or temporary facility change, procedure change, or test or experiment for the purpose of determining if it involves any one or more of the following:

1) Changes to the Technical Specifications or Operating License

2) Changes to the facility as described in the USAR

3) Changes to procedures as described in the USAR

4) Tests or experiments not described in the USAR

If the proposed action requires a change to the Technical Specifications or Operating License, a significant hazards consideration is performed. The significant hazards consideration is prepared in accordance with procedure NG-NS-00801, "Operating License Amendments". NRC approval of the change is required before the proposed change could be implemented.

If the safety review determined that the change involved any of items 2 through 4 above, the second tier process is initiated by performing a 10 CFR 50.59 safety evaluation. This safety evaluation is a written technical evaluation that provides the bases for the determination that the proposed change, test, or experiment does or does not involve an unreviewed safety question.

Individuals that prepare, review, and approve safety reviews/evaluations have completed a 10 CFR 50.59 training program and are designated in writing as qualified by their respective department Director. In addition, the safety review/evaluation reviewer and approver are qualified in accordance with procedure NG-VP-00132, "Qualified Reviewer Program". The Qualified Reviewer Program establishes personnel qualification requirements in accordance with ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel".

Safety evaluations are also approved by the onsite Station Review Board prior to implementation and are subsequently reviewed by a subcommittee of the off-site Company Nuclear Review Board (CNRB) for additional assurance that the facility or procedure change is adequately addressed and that no unreviewed safety question exists.

Facility changes that involve an unreviewed safety question are submitted to the NRC for approval prior to implementing the proposed change. This process is governed by procedure NG-NS-00801, "Operating License Amendments".

A.6.0 Updated Safety Analysis Report (USAR) Change Process

The process that implements 10 CFR 50.71(e) at the DBNPS is governed by procedure NG-NS-00806, "Preparation and Control of USAR Changes". This procedure also provides configuration control for changing documents that are incorporated by reference into the USAR, such as the "Fire Hazards Analysis Report" (FHAR) and the "Technical Requirements Manual" (TRM).

The FHAR contains the description of the DBNPS fire protection program and is incorporated by reference into the USAR. The TRM contains selected requirements which have been relocated from the Operating License Technical Specifications and is also incorporated by reference into the USAR.

These items were relocated to the TRM based on the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Plants, and 10 CFR 50.36, "Technical Specifications," as amended in Final Rule published in the Federal Register dated July 13, 1995. The TRM was developed to centralize the requirements relocated from the Technical Specifications and provide for the necessary administrative controls.

A.6.1 10 CFR 50.71 Submittal

Changes to the USAR are identified via the modification process, by procedure changes, by correspondence (including License Amendments), by the USAR review effort prior to each submittal, or by individuals reviewing the USAR. Once the potential change is identified, a Safety Review/Evaluation is performed. The USAR change is reviewed by the initiator's management, followed by a review of the department responsible for the information in the affected USAR section. The proposed change is also reviewed by the DBNPS Regulatory Affairs Section to determine if other sections of the USAR are affected by the change.

Those changes to USAR Section 17.2, "Quality Assurance During The Operations Phase", also receive a review in accordance with 10 CFR 50.54(a). If the USAR change is determined to be a reduction in commitment to the "Nuclear Quality Assurance Program" as described in USAR 17.2, the change is submitted to the NRC for approval prior to implementation.

The DBNPS USAR is updated and submitted to the NRC six months after each refueling outage in accordance with 10CFR 50.71(e). The USAR is used on a day-to-day basis to support plant operations (e.g., to perform safety reviews/evaluations). To provide for a current description of the facility, USAR changes are "posted" against the USAR once the USAR change is approved. Following completion of the facility change (ie., a system returned to service or a procedure made effective) the USAR change is statused "as-built", along with the affected procedures and drawings. This provides for a current facility description for personnel performing 10 CFR 50.59 safety reviews/evaluations.

Prior to the 10 CFR 50.71(e) submittal, proposed USAR changes are submitted to the DBNPS Interdisciplinary Review Group (IRG) for review. The IRG is a multi-disciplinary onsite group consisting of experienced representatives from Engineering, Nuclear Assurance, Regulatory Affairs, Operations, and Independent Safety Engineering who are very knowledgeable of the USAR. Changes to the USAR require a safety review/evaluation, an existing NRC Safety Evaluation Report (SER), or NRC approved 10 CFR 50.54(a) evaluation (for QA program changes) which adequately justifies the proposed change.

A.7.0 Program Assessments

An assessment of the "Nuclear Quality Assurance Program" and site activities, such as design control, configuration control, and procedure control is conducted through scheduled audits performed under the oversight of the CNRB. This includes periodic audit participation by CNRB members and external organizations, such as the Institute of Nuclear Power Operations (INPO) and the Joint Utility Management Assessment (JUMA) team. Audits are performed by Quality Assessment, as a minimum, in accordance with the frequency specified in the Technical Specifications. These audits assess the performance of activities required by the "Nuclear Quality Assurance Program". Additional assessments (surveillances) are performed based on current events, trending analysis, or management/QA concerns.

NRC Request for Information:

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

Toledo Edison Response:

B.1.0 Course of Action Program

The DBNPS "Course of Action Program" following the June 9, 1985 loss of feedwater event, established a configuration management program to provide equipment/system data bases and to develop and maintain this documentation. The "System Review and Test Program" (SRTP) identified the system functions required for safe operation. The SRTP program evaluated the scope of the surveillance testing and reviewed this testing for completeness and adequacy.

Additionally, maintenance procedures were strengthened by the maintenance improvement program.

The "Configuration Management Program" developed two documents that capture design bases information, the Design Criteria Manual and System Descriptions.

These efforts are described in detail below.

B.1.1 System Review and Test Program

The DBNPS established the "System Review and Test Program" (SRTP) to address concerns about the adequacy of safety systems and engineered safety features at the DBNPS. Specifically, the program considered the concerns related to:

1) the adequacy of safety system testing, including verification that safety systems were tested in configurations required by design-basis analysis; and

2) the adequacy of engineered safety features, including design considerations.

The SRTP was designed to provide a comprehensive evaluation of selected systems. The system evaluation was intended to identify and resolve concerns and specify required testing to verify the ability of the system to perform its intended function. The conclusion of the system evaluation would verify that the system would function as designed.

The systems selected for review in the SRTP were those systems deemed to have the most impact on the safe operation of the DBNPS. The following 34 systems were within the scope of the SRTP:

Reactor Coolant; High-Pressure Injection; Core Flooding; Decay Heat Removal and Low-Pressure Injection; Containment Spray; Containment (Station) Emergency Ventilation; Containment Air Cooling and Hydrogen Control; Make-up and Purification; Electrical 125/250-V dc (including battery room heating and ventilation); Electrical 4.16-KV (13.8/4.16-KV transformers); Electrical 480V distribution (including inverters and required transformers); Electrical 13.8-KV (including startup and auxiliary transformers); Emergency Diesel Generators (including "Q" fuel oil tanks and diesel room ventilation); Instrument AC Power (including inverters and required transformers); Anticipatory Reactor Trip; Control Rod Drive Control, Incore Monitoring (including core exit thermocouples); Reactor Protection; Steam and Feedwater Rupture Control; Safety Features Actuation; Integrated Control/Non-Nuclear Instrumentation; Security; Control Room Normal and Emergency Heating and Ventilation; Station and Instrument Air; Station Fire Protection; Component Cooling Water; Service Water; Auxiliary Feedwater; Main Steam; Steam Generator; Main Feedwater; Gaseous Radwaste; Post-Accident Sampling; and Miscellaneous Containment Isolation Valves.

The SRTP reviewed the history of the systems to:

1) identify important and recurring design, maintenance, and operational problems and determine the corrective actions required;

2) evaluate the scope of existing periodic testing and identify additional testing needed to ensure system operability;

3) conduct a test program to ensure the system functions as intended;

4) verify the adequacy of modifications completed during the outage following the June 9, 1985, event.

The system reviews consisted of the identification of functions stated in the Operating License Technical Specifications (TS), the Updated Safety Analysis Report (specifically USAR Chapter 15 and the applicable USAR sections for safety functions), the NRC's Safety Evaluation Report (NUREG-0136) for the DBNPS Operating License, and plant procedures.

Following these reviews, restart tests were developed and performed validating that the systems were capable of performing their intended functions. The restart testing program concluded with a demonstration of plant, and system response from a trip of one main feedwater pump and the evaluation of a trip of the unit from high power to determine that systems would respond as required.

B.1.2 Configuration Management

As part of the Davis-Besse Course of Action Program, the Configuration Management Program was established to document the design basis and configuration of the plant. The objectives of the Configuration Management Program were to:

1) determine the as-built configuration of the plant by conducting system walkdowns to collect equipment information;

2) use the equipment data to validate vendor and Toledo Edison documentation;

3) use the equipment data to help establish System Descriptions which would be used for detailed design work and for the support of station activities;

4) establish a "living" data base; and

5) develop procedures to maintain validated data base information current.

In meeting these configuration management objectives, the following products were developed: System Descriptions; Operational Schematics; Davis-Besse Configuration Equipment Summary (DBCES); and Design Criteria Manual (DCM).

B.1.2.1 Design Criteria Manual

The "Design Criteria Manual" (DCM) delineates the basic criteria applied to the plant design, general conditions under which the plant and its systems are required to perform, and summarizes the governing codes and regulations applicable to the plant. It is intended that the DCM be used to provide a general understanding of the plant design bases and the relationship between its component parts.

The "Design Criteria Manual" covers such things as event scenarios and protection from natural phenomena, missiles, and postulated piping failures. It provides basic criteria for design requirements such as fire protection, seismic and environmental qualification, and it delineates requirements such as separation criteria and single failure analysis. The DCM provides general design information for major plant structures, security, electrical, control, heating ventilation and air conditioning (HVAC), and piping.

B.1.2.2 System Descriptions

"System Descriptions" (SDs) serve as a "road map" to understand the design basis of plant systems. The SDs include the system design basis, the system and key component engineering requirements, the system and key component functions, regulatory requirements, operational phases, Technical Specifications references, and testing and surveillance requirements.

The "System Descriptions" cover the Maintenance Rule Risk Significant Systems along with the majority of the remaining plant systems. Although developed prior to the establishment of industry standards, the SDs meet the intent of NUMARC 90-12, "Design Basis Guidelines". The SDs are controlled and maintained by procedure EN-DP-01150, "System Descriptions", and are available to plant personnel as a primary resource in obtaining design information.

The SDs were developed by reviewing information from: the USAR, the Technical Specifications, applicable NRC regulatory requirements, system and plant procedures, component specifications, calculations, test results, startup test files, records management system files, the DBNPS SRTP Reports, equipment information obtained from configuration management walkdowns, and other appropriate sources to assist in documenting the requirements and design of that system.

B.1.2.3 Maintenance Improvement Program

The Maintenance Department administrative and technical procedures underwent a major upgrade during 1985 and 1986. The objectives of the procedure upgrade were to:

1) incorporate guidance into administrative procedures derived from the NRC, INPO and other pertinent industry practices;

2) incorporate previous maintenance experience in technical procedures;

3) establish better defined administrative and work controls for plant maintenance activities; and

4) assure that formalized feedback mechanisms were established that would improve the quality and accuracy of the technical procedures as more experience was gained.

Additionally, maintenance procedures were reviewed for incorporation of vendor recommendations and guidance.

B.2.0 Revalidation of Selected Operations Procedures

Toledo Edison performed an audit of DBNPS procedures from September 6, 1988, through October 14, 1988. The findings and recommendations resulting from this audit led to a corrective action of revalidating selected operations procedures prior to startup from the Fifth Refueling Outage (5RFO). The Operations procedures validation and reverification effort was implemented on approximately 40 key operational procedures. The procedures that were revalidated were identified by the Operations Section, supplemented and concurred with by the Independent Safety Engineering Unit, and approved by the Plant Manager. Procedures found to require revisions were revised prior to entering Mode 2 (startup) following 5RFO.

B.3.0 Design Control Process

Changes to the plant design are captured and translated into the operating, maintenance, and testing procedures by the DBNPS modification process as discussed in response to Item (a). This process provides the mechanisms for changes to design bases requirements to be translated into operating, maintenance, and testing procedures by implementing such actions as:

1) Implementing interfacing procedure changes with the modification;

2) Performing post modification testing to demonstrate that the change was implemented in accordance with the modification's design;

3) Revising vendor manuals to reflect as-built information;

4) Revising drawings, specifications and other design output documents to reflect installed configuration;

5) Completing appropriate training on plant systems prior to returning equipment back to service.

B.4.0 Internal Assessments

Further assurance for concluding that design bases requirements are translated into operating, maintenance, and testing procedures is provided through internal assessments and audits. The DBNPS Independent Safety Engineering (ISE) Unit has conducted a number of significant reviews, including several safety system functional-type inspections (SSFIs). These reviews were "vertical slice" reviews modeled after the NRC's technique for conducting such audits as described in NRC Inspection Manual Procedure 93801, "Safety System Functional Inspection."

B.4.1 Independent Safety Engineering Reviews

The following significant "vertical slice" reviews have been performed since 1989 by the ISE Unit:

1) The Station and Instrument Air System SSFI was conducted in 1989. Selected supporting systems, as well as certain end user components supplied by the air system were reviewed. In addition, the EDG Air Start System was reviewed. The review primarily focused on the following functional areas: design/design changes, maintenance, operations, testing, and training. An additional area, the application of industry operating experience, was a secondary focal point and was reviewed in regard to its interface with the five primary areas assessed.

   The overall assessment of the Station and Instrument Air System was that the system was capable of performing its intended function. The assessment also concluded that generally there were adequate programmatic controls in place regarding the areas of design, maintenance, operation, testing, and training. The majority of the observations related to the end user components (e.g., air operated valves) rather than the air system itself. Observations from this review were tracked and appropriately dispositioned.

2) The 1991 an Emergency Diesel Generator (EDG) SSFI was conducted to determine whether the EDGs and support systems were capable of performing their intended safety function as designed, configured, and installed. The team reviewed EDG electrical and mechanical system designs, including supporting calculations, operations, maintenance, testing, and procedures.

The team considered the most notable areas in need of improvement to be: 1) instances where all supporting calculations or engineering justifications did not exist; and 2) existing calculations did not appear to fully address all aspects necessary to adequately capture the design basis. In general, based on the available information and the experience of the reviewers, the team judged that the EDGs and their support systems were designed, configured, maintained, and operated in a manner such that they would satisfactorily perform their intended safety function. Specific weaknesses identified during this review were tracked and appropriately dispositioned.

3) A Steam Generator SSFI was conducted in 1992. The purpose of the SSFI was to assess the operational readiness of the Steam Generators (SG) to perform their functions important to plant safety. In addition, interfacing system attributes, such as flow rates, setpoints, maintenance, and material condition were reviewed as they may impact the SGs. These interfacing systems included: Auxiliary Feedwater, the Main Steam Isolation Valves, and the Main Steam Non-Return Valves.

   The overall conclusion from this review was that the steam generators were capable of performing their functions important to plant safety. The assessment found no design changes that adversely affected or violated design requirements. The team found that the ability to retrieve design and design basis information was a strength. Opportunities for improvement were identified in the areas of System Description maintenance and the main steam line break analysis of calculated pressures in the main steam line room. Specific weaknesses identified during the review were tracked and appropriately dispositioned.

4) The Service Water Safety System Functional Review (SW SSFR) was performed in 1993 to assess the operational readiness of the Service Water System (SWS). Compliance with the design and operational requirements of the SWS were reviewed using information from the USAR, System Descriptions, Technical Specifications, vendor manuals, instrument data packages, "Design Criteria Manual", applicable plant drawings and procedures, and NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment".

   Among the issues identified, the most notable item concerned SWS calculations. The existing SWS calculations did not provide sufficient detail to verify that the assumptions and conclusions made were consistent with the current system configuration. In response to PCAQRs generated during this review, several SWS calculations were revised.

   The SSFR concluded that the Service Water System would perform its nuclear safety-related function. Potential Conditions Adverse to Quality Reports (PCAQR) and ISE open items were issued to track issues to completion. No equipment functionality concerns were identified during this assessment.

5) The System-Based Instrument and Control Inspection (SBICI) conducted in 1994, was based on guidance found in NRC Inspection Procedure 93807, "System Based Instrumentation and Control Inspection". The purpose of the inspection was to verify the functionality of selected instrument loops or strings. The selection of systems and instrument loops was based on evaluation of the Davis-Besse Individual Plant Examination (IPE), plant personnel interviews, regulatory input, and input from knowledgeable individuals.

   No functionality concerns were identified. The inspection team concluded that, in general, the Instrument and Control (I&C) systems and their respective loops were capable of performing their intended functions. However, programmatic issues were identified in the areas of uncertainty calculations and channel functional test performance. These issues were tracked and appropriately dispositioned.

6) During 1994 and 1995, Engineering and ISE embarked on a joint project to perform a comprehensive assessment of the Auxiliary Feedwater (AFW) System functions. This effort was undertaken for the following reasons: AFW System failures are significant contributors to the core damage frequency in the DBNPS Probabilistic Risk Assessment (PRA); numerous changes have been made to this important system; and issues and questions related to the AFW System design basis, such as LER 93-007, "Plant Operated Outside Design Basis Due to Isolation of No. 1 Steam Generator", dated December 13, 1993.

   This project assessed the AFW function by incorporating elements of NUMARC 90-12, "Design Basis Program Guidelines", SSFIs, and Davis-Besse's SRTP. Systems and components reviewed included the: steam supply to the AFW Pump Turbines (AFPT); AFW pumps; AFPTs; the AFW discharge to the SGs; active and passive components within the steam and water flow paths; the AFW SG level control system; and the AFW functions of the Motor Driven Feedwater Pump (MDFP).

   The overall results of this assessment concluded that the AFW System is capable of performing its intended design function. Several improvements were made in the understanding of the AFW System Technical Specification bases and testing requirements. The USAR was also updated regarding commitments to 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) events," and Chapter 15.2.8, "Loss of Normal Feedwater". Where necessary, PCAQRs and ISE open items were generated to track and disposition issues.

   The detailed review of component design functions associated with design basis has not been completed. Additionally, the project recognized that the level of detail documenting the assumptions made in the original supporting calculations should be enhanced and validated. These issues have been turned over to Engineering for follow-up review and inclusion in the Design Basis Validation Program.

B.4.2 Engineering and Quality Assurance Assessments

The following significant Engineering Assurance and Quality Assurance assessments have been performed:

1) In 1989 DBNPS Engineering Assurance performed several "vertical slice" design evaluations for the purpose of providing assurance that the design basis inputs of selected modifications had been appropriately defined and implemented in lower level documents. Concerns identified during these evaluations served to raise engineering management's awareness of the need for more careful checking of the engineering product before issuance.

2) In late 1993, Quality Assurance identified an increasing trend of design/verification deficiencies when five PCAQRs were initiated in one month in this area. This trend generated "Suspected Trend Investigation Report" (STIR) 93-002. The follow-up investigation concluded that no special, single cause was associated with the five events identified in the PCAQRs. The deficiencies had existed for a long period of time with no adverse effect on programs, personnel, or hardware. The individual deficiencies were addressed and corrected by their respective PCAQRs and proposed corrective actions were considered appropriate.

3) A review of QA audits and surveillances conducted over the last six years indicate that engineering programmatic deficiencies have been identified (configuration and modification control activities). Appropriate corrective actions have been initiated and completed to resolve the identified concerns.

4) Recent trending of corrective action documents (PCAQRs) has not indicated any adverse trends in discrepancies between the station configuration and critical design output documents. This is an indication that design bases requirements are being properly translated to the field. The DBNPS QA trending program has identified an increased number of discrepancies between the as-built facility and the USAR. As described in response to Item (e), a root cause evaluation of the discrepancies identified during the recent USAR review is being performed.

B.5.0 External Assessments

In addition to the previously described internal reviews, NRC inspections have been conducted in recent years, such as the Special Electrical Distribution System Functional Inspection (EDSFI) and the Service Water System Operational Performance Inspection (SWSOPI). These inspections concluded that the systems reviewed were capable of performing their intended functions. These inspections did identify discrepancies and weaknesses for which appropriate corrective actions were taken. A Configuration management concern was identified during the SWSOPI. This concern was addressed as discussed in response to Item (c), specifically section C.1.2.

B.6.0 Conclusion

The DBNPS internal assessments concluded that the safety systems were capable of performing their intended functions. The assessments also concluded that generally there were adequate programmatic controls in place regarding the areas of design, maintenance, operation, testing, and training. The reviews identified a common weakness in some design calculations such as: calculations could not be found to support various aspects of design basis, and calculations did not always provide sufficient detail to verify that assumptions and conclusions made were consistent with current system configuration.

As previously discussed in the cover letter, the DBNPS is undertaking a Design Basis Validation Program to validate the existing system design bases, with emphasis being directed to reviewing design basis analysis and supporting calculations. Current practice, as described in response to Item (a), requires a detailed review of existing calculations as part of the modification process to confirm calculational assumptions.

Additional assurance for concluding that design bases requirements are translated into operating, maintenance, and testing procedures is provided in response to Item (c). For example, station personnel use design output documents (e.g., P&ID's) to verify configuration control.

NRC Request for Information

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

Toledo Edison Response

The response to this request requires information which can be most clearly presented by discussing the issues of configuration and performance separately.

C.1.0 Configuration Consistent with Design Bases

C.1.1 System and Component Configuration

Functions important to the safe operation of DBNPS were identified as part of the System Review and Test Program (SRTP). "Systems Descriptions" (SDs) were developed to provide a comprehensive summary of the system design basis technical documents that describe the design, operation, maintenance, and reliability requirements of individual plant systems.

The SDs describe system functions and contain the design requirements and the design requirement bases for each system. The System Descriptions are available to plant personnel and can be used as a primary resource in obtaining design basis information. Complimentary to SDs, DBNPS Piping and Instrument Diagrams (P&ID) and Operational Schematics (OS) are design output documents which depict the normal system alignments.

The OSs contain a compilation of system information and provide a quick reference for system operation related details. Because the plant was designed and normally operates at or near full rated power, the P&IDs and OSs depict the normal system alignments for full power operation in accordance with procedure NG-EN-00307, "Configuration Management". Systems which do not have specific required normal lineups at full power are depicted on the drawings in accordance with procedure NG-EN-00307, "Configuration Management".

For systems which have specific lineups at full power, system operating, maintenance, and test procedures are based on maintaining and/or restoring systems to the P&ID and OS depicted lineups. This provides reasonable assurance that components are aligned in a known, analyzed configuration, which supports the design bases.

When the plant is not operating at full power, the need to maintain the necessary design bases safety functions is also recognized at DBNPS. System alignments during these times are specified in the system operating procedures. An evaluation of changes to system operating procedures is conducted in accordance with 10 CFR 50.59 to review the change for an unreviewed safety question.

Before a planned shutdown, a schedule is developed and thoroughly reviewed in compliance with procedure NG-DB-00116, "Outage Nuclear Safety Control". This procedure invokes requirements above and beyond the requirements specified in the DBNPS Technical Specifications and provides a defense-in-depth approach to maintaining plant safety. This defense-in-depth philosophy is further implemented by procedure DB-OP-06904, "Shutdown Operation". The required key safety functions and the availability of the systems that support each function are tracked and posted via a color-coded report on a daily basis. Equipment which is relied upon to provide the design basis safety functions is designated as "Protected Train" equipment and identified by placards throughout the plant. Activities which could affect "Protected Train" equipment are restricted. In summary, these actions are taken to provide assurance that plant configuration during shutdown is controlled in a manner such that design bases safety functions are available.

During normal full power operation, for those systems with specific required lineups, the applicable USAR, OSs, P&IDs, and Operating Procedures specified system configuration are in agreement and match the actual component positions in the plant for normal system operations. An NRC Service Water System Operational Performance Inspection (SWSOPI) performed in late 1993 (Inspection Report 93-016) identified instances where the documentation and/or field positions were inconsistent with this requirement. Investigations into these discrepancies led to the issuance of Management Corrective Action Report (MCAR) 93-002. The MCAR provided a complete evaluation of such inconsistencies for extent of condition and corrective actions. This MCAR has been dispositioned as described below.

C.1.2 Configuration Control Improvements

Management Corrective Action Report 93-002 resulted in a number of corrective actions which led to improvements in configuration control. The most significant corrective actions were as follows:

1) Plant systems were walked down. The USAR, P&ID, OS, and system operating procedure specified system alignments were compared against each other and the field positions. The P&ID and OS functional representation of the as-built system was verified to be correct. Identified discrepancies were corrected.

2) The Configuration Management (CM) process, procedure NG-EN-00307, "Configuration Management", and the procedure change process, procedure NG-NA-00115, "Control of Procedures", were revised to require concurrent revision of critical CM documents and procedures. This helps to ensure that procedures and critical design documents match the as-built plant, and that Design Engineering reviews the changes to normal system alignment prior to implementation. Critical design documents are those that are located in the control room to support Operations in day-to-day activities.

3) The controls for temporary system alignment changes were improved. Temporary alignments to safety related equipment are immediately evaluated for impact on operability in accordance with Generic Letter 91-18, "Information to Licensees Regarding Two Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability", and Operations Policy GP-21, "Principles for Dealing with Operability Issues". Changes were implemented to procedures used to temporarily align equipment (i.e., procedure DB-OP-00015, "Safety Tagging"; and procedure DB-OP-00016, "Removal and Restoration of Station Equipment"). The changes provide reasonable assurance that temporary alignments are not used in lieu of procedure alterations and that 10 CFR 50.59 evaluations are performed if maintenance does not allow restoration of non-safety related equipment to the normal alignment in a reasonable period of time.

4) Procedure DB-DP-00307, "Station Configuration Control", was developed to provide controls over positionable components. Positionable components are defined as components which are manipulated as a normal part of system operation to control the flow of fluids (including steam and gas) or electricity. This new procedure accomplished the following:

   a) Established controls over service connections such as demineralized water, station air, and lighting panels;

   b) Delineated responsibility for operation of plant components between Operations, Maintenance, and Chemistry;

   c) Improved component labeling conventions;

   d) Provided a method to ensure that positionable components are repositioned if a procedure change is issued which specifies a new component position applicable to the current mode of operation.

C.1.3 Internal Assessments

The adequacy and effectiveness of the corrective actions discussed above have been assessed several times since their implementation. Some of these assessments identified opportunities for additional improvements, such as in the area of chemistry sampling valves. However, in general, these assessments concluded that the corrective actions were adequate in scope and are being effectively implemented. This has been recently illustrated by a Quality Assurance audit of design engineering functions (Audit AR-96-DESIN-01 dated July 29, 1996), which concluded that adequate programmatic controls exist and are being implemented effectively.

C.1.4 Equipment Walkdowns

There were several additional significant initiatives performed at Davis-Besse related to verifying that a SSC configuration is known, correctly documented, and consistent with the design bases:

1) In the early and mid-1980s, there were extensive walkdowns performed to address concerns raised in NRC I&E Bulletin 79-02, "Pipe Support Baseplate Design Using Concrete Expansion Anchorbolts", and NRC I&E Bulletin 79-14, "Seismic Analysis for As-built Safety Related Piping Systems". These walkdowns verified actual piping configuration, and the adequacy of piping supports, baseplates, and anchor bolts.

2) In the mid-1980s, detailed inspection plans were developed to walkdown fire barriers. The walkdowns verified or established accurate configuration information for fire dampers, fire doors, structural steel fireproofing, as well as fire and flood barriers. The walkdown data was used to develop detailed drawings, calculations, and procedures.

3) In the mid-1980s, extensive walkdowns were performed on equipment required to be Environmentally Qualified (EQ) in accordance with 10 CFR 50.49, "Equipment Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants". The walkdowns were performed to verify the configuration of EQ equipment and verify that it was qualified for its location/application. Data from these walkdowns was also utilized in developing the DBNPS' equipment data base known as DBCES.

4) Plant equipment walkdowns were conducted in the late-1980s as part of establishing a configuration management program. The walkdowns were done in accordance with the Configuration Management Program Manual to verify that accurate information on installed plant equipment was captured and correctly depicted on design drawings. The plant's CM equipment data base (DBCES) was developed based on the information from these walkdowns. The DBCES data base provides information at the component level and is maintained as part of the design control process as discussed in Section A.3.0, "Configuration Control Process".

C.1.5 Structures Configuration and Maintenance of Structures

Management Corrective Action Report 93-002 and the associated corrective actions focused heavily on systems and components. Engineering also implemented a program to provide reasonable assurance that structures are configured and maintained consistent with the design basis. As part of the DBNPS Maintenance Rule Program, Engineering performs civil/structural reviews of structures at Davis-Besse. These evaluations are conducted in accordance with 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants". The structures included within the scope of these evaluations are:

1) Safety related structures

2) Non-safety related structures

   a. that are relied upon to mitigate accidents or transients; or

   b. that are used in plant emergency operating procedures (EOPs); or

c. whose failure could prevent safety related SSCs from performing their safety-related function(s); or

d. whose failure could cause a reactor scram; or

e. whose failure could cause actuation of a safety related system.

C.1.6 USAR Review

A USAR review and update was completed and documented in DBNPS transmittal letter to the NRC, Serial 2423, dated December 2, 1996. This review was performed to provide reasonable assurance that the USAR accurately describes the facility design and its operation. Although there were 125 USAR changes generated as a result of the review, only one discrepancy was found to affect equipment operability. This condition was reported in Licensee Event Report (LER) 96-007, "Control Room Emergency Ventilation System Design Basis Calculation Error". From this effort, it is concluded that the USAR does, with reasonable accuracy, depict facility design and operation. This review is further described in the response to Item (e).

C.1.7 Ongoing Assessments

The Operations staff routinely monitors the configuration of components in the plant providing assurance that they are properly positioned. Plant Operations employs a number of methods as detailed below:

1) Procedure DB-OP-04004, "Locked Valve Verification": This quarterly test checks the position of approximately 250 key components controlled by the Locked Valve Program described in procedure DB-OP-00008, Operation and Control of Locked Valves. Valves are placed in this program and locked if the position of the valve is essential in supporting the intended safety function of the system or to meet a design code or regulation. These valves are typically located in standby systems such that mispositioning of the valve may not be detected during normal plant operation. This verification is also performed prior to mode changes as described in procedure DB-OP-06911, "Prestart-up Checklist".

2) Procedure DB-OP-04005, "Capped Valve Verification": This monthly test checks the position of approximately 200 key components controlled by the Capped Valve Program described in procedure DB-OP-00009, "Operation and Control of Capped Valves". In general, valves are placed in this program if the position of the valve is essential in supporting containment integrity or plant safety. This verification is also performed prior to mode changes as described in procedure DB-OP-06911, "Prestart-up Checklist".

3) A number of plant tours are completed by operators and supervision each day. Guidance for these tours is provided in procedure DB-OP-00005, "Operator Logs and Rounds". This procedure provides direction for general area inspections and equipment checks.

These inspections and checks verify equipment condition and configuration are normal. Abnormal conditions are reported, investigated, and resolved.

4) Selected parameters are monitored using plant readings. Many of these readings are taken to verify that key system conditions or configurations are being maintained. Abnormal conditions are indicated by audible and/or visual cues. These conditions are promptly reported, investigated, and resolved.

5) Electrical breaker positions of molded case 480 VAC and high voltage breakers are verified for proper position at least once per day during operator rounds in accordance with procedure DB-OP-00005, "Operator Rounds and Logs".

6) Plant Computer alarms provide a visual indication in the control room. Alarms are used to indicate that key system conditions or configurations are not as expected. Alarms are promptly reported, investigated, and resolved.

7) Plant Annunciator alarms provide a visual and audible indication in the control room. These alarms are used to indicate that key system conditions or configurations are not as expected. Alarms are promptly reported, investigated, and resolved. Specific procedural direction is provided for each annunciator window.

8) System Status Files provide a method to document the current configuration for each system/component. These files are controlled in accordance with procedure DB-OP-00000, "Conduct of Operations".

C.1.8 Conclusion

In summary, Davis-Besse has performed extensive reviews and walkdowns, and has adequate programs in place to provide reasonable assurance that SSCs are configured consistent with the design bases. The documents and data bases which include configuration information or provide direction related to plant operation are maintained consistent with the as-built plant design basis in accordance with the Configuration and Design Control processes, as previously discussed in response to Item (a).

C.2.0 Performance Consistent with Design Bases

The second aspect of this request was to provide the rationale for concluding that SSC performance is consistent with the design basis. The verification of adequate performance is primarily established through various testing programs.

C.2.1 System Test Program

The most significant of these programs is the Surveillance and Periodic Test (ST/PT) Program, which is conducted in accordance with procedure NG-DB-00202, "Test Control." This program was verified and modified to provide reasonable assurance that adequate testing was included to ensure that systems would perform their intended functions in configurations required by the design basis. The ST/PT test program requirements have been maintained consistent with the design basis. The ST/PT program includes the testing necessary to satisfy Technical Specification requirements (STs) as well as testing to verify functions which support or could challenge the performance of safety systems (PTs).

C.2.2 Equipment Test Program

Complimentary to the ST/PT program are a number of other testing programs which verify SSCs will perform in accordance with the design basis:

1) To supplement the ASME Section XI Inservice Testing Program in providing reliable indication of motor operated valve (MOV) capabilities, a "MOV Program Manual" was established to test MOVs under design basis conditions. This program was established in accordance with the requirements of Generic Letter (GL) 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance". The GL 89-10 program at DBNPS has been inspected by the NRC, as documented in NRC letter "Close-out of NRC Review of Generic Letter 89-10; Safety Related Motor-Operated Valves Testing and Surveillance Program," dated October 17, 1996.

2) The ASME Section XI Inservice Inspection/Inservice Test (ISI/IST) Program is conducted in accordance with procedure NG-EN-00314, "Inservice Inspection". This is an integrated program of inspections, examinations, and tests implemented to provide reasonable assurance that the structural and pressure-retaining integrity of components and that the operational readiness of pumps and valves in accordance with the requirements of ASME Section XI and the DBNPS Technical Specifications, is maintained.

3) A Preventive Maintenance (PM) Program is implemented in accordance with procedure DB-PF-00002, "Preventive Maintenance", and includes checks of various component calibrations to provide reasonable assurance these components will respond when required.

C.2.3 Plant Reliability Test Program

Other programs have been established to enhance plant reliability and in many cases support those programs discussed above. For example:

1) A check valve reliability program was established in response to concerns identified in INPO Significant Operating Experience

Report (SOER) 86-03 "Check Valve Failures or Degradation" dated October 15, 1986. This program prioritizes check valve inspections and testing based on functions required and system modeling. The periodic inspections are conducted in accordance with the PM program and results are fed back into the PM program to improve reliability. In some cases, diagnostic testing was performed to provide information on check valve performance.

2) A predictive maintenance program is implemented in accordance with procedure EN-DP-01074, "Predictive Maintenance", and provides benefits such as identifying degrading performance of various equipment and components prior to required action levels. For example, infrared thermography, vibration analysis, and lubrication analysis provide indication of component degradation prior to component failure.

3) A "Post Modification and Maintenance Test Program" is in place to provide for appropriate testing of SSCs following maintenance or modification activities. MODs are reviewed and testing is specified in accordance with procedure EN-DP-01072, "Modification Test Requirements". Maintenance activities are reviewed and testing is specified in accordance with procedure DB-PF-01025, "Maintenance Testing Requirements". A Post Maintenance Test Manual was developed to help identify applicable testing requirements for SSCs following maintenance or modification. The manual is implemented in accordance with procedure DB-DP-00007, "Control of Work". Utilization of this manual privides additional assurance that the appropriate testing is performed to confirm continued SSC function per the design bases.

4) A "Corrosion-Erosion Monitoring and Analysis Program" (CEMAP) was developed in accordance with Generic Letter 89-008, and is implemented by procedure EN-DP-01301, "CEMAP". The program is designed to provide adequate prediction and monitoring of piping such that wall thinning below design allowables does not occur in systems subject to significant erosion/corrosion.

C.3.0 Owners Group Reviews

The DBNPS staff also actively participates in a number of owners group committees. Many of the projects and programs overseen by the committees are related to ensuring that SSCs perform in accordance with the design bases. A few examples are:

1) Control Rod Drive Life Extension

2) Excore Dosimetry Monitoring/Reactor Vessel Integrity

3) Pressurizer Surge Line Thermal Stratification

4) Steam Generator Tube Plugging

5) Control Rod Drive Mechanism (CRDM) Nozzle Inspection

C.4.0 Conclusion

In summary, the testing programs at DBNPS have been adequately implemented, verified, and performed to provide reasonable assurance that SSCs will perform as required to support the design bases. Testing procedures and programs are maintained consistent with the as-built plant design basis in accordance with the Configuration and Design Control processes discussed in response to Item (a).

NRC Request for Information:

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

Toledo Edison Response:

D.1.0 10 CFR 50 Appendix B

The Davis-Besse Updated Safety Analysis Report (USAR) Sections 17.2.15, "Nonconforming Materials, Parts, or Components", and 17.2.16, "Corrective Action", requires in part that procedures be established to ensure that conditions adverse to quality are promptly identified, documented, evaluated for their significance, and corrected. Additionally, the USAR requires that corrective action documents contain provisions for identifying the root cause of the condition adverse to quality, completing remedial corrective actions, and corrective actions to prevent recurrence. Further, the Davis-Besse corrective action programs were designed to meet 10 CFR 50 Appendix B Criterion XV, "Nonconforming Materials, Parts, or Components"; Criterion XVI, "Corrective Action"; Criterion XVIII, "Audits"; ANSI N45.2.12-1977, "Requirements for Auditing Quality Assurance Program for Nuclear Power Plants"; and applicable USAR requirements. These corrective action document requirements are then further defined in the Nuclear Quality Assurance Manual (NQAM), ASME Quality Assurance Manual (AQAM) and Nuclear Group (NG) procedures.

Changes to these programs are reviewed in accordance with 10 CFR 50.59 and 50.54(a) as necessary.

The primary corrective programs at Davis-Besse Nuclear Power Station (DBNPS) are implemented by:

1) Procedure NG-NA-00702, "Potential Condition Adverse to Quality Reporting", and

2) Procedure NG-NA-00701, "Nuclear Assurance Audit and Surveillance"

Both processes are managed by the Nuclear Assurance Department and administered in accordance with the referenced Nuclear Group Procedures.

D.2.0 Potential Conditions Adverse to Quality

Prior to 1986, several processes existed to document conditions adverse to quality outside the audit process. These included nonconformance reports, deviation reports, supplier deviation reports, and stop work actions. Since 1986, the DBNPS corrective action program has shifted to a single program to document these types of nonconforming conditions, as well as potential conditions adverse to quality.

The Potential Condition Adverse to Quality Report (PCAQR) process is the primary method of reporting and documenting any condition that an individual believes is potentially adverse to quality. This process is used to improve plant operation and safety by identifying issues, observations, or concerns with hardware, programs, or activities regardless of quality classification. The PCAQR process involves every organization at the DBNPS.

Previously, significant conditions adverse to quality were addressed on a Management Corrective Action Report (MCAR). Currently, conditions adverse to quality that are considered significant or that represent an adverse trend would be addressed as a Category 1 or 2 PCAQR. This document would receive Director level involvement, formal root cause evaluation, and follow-up review or audit to verify the adequacy of the corrective actions.

The PCAQR is evaluated in six distinct and definite steps:

1. Initiate PCAQR - Anyone may document/identify potential problems. Identification of potential problems may come from involvement in day-to-day operations, maintenance, or engineering. Additionally, self-assessment programs such as the Maintenance Continuous Performance Monitoring program may result in the initiation of a PCAQR.

   The PCAQR is then submitted to Operations for prompt review. This review includes immediate determination of reportability to the NRC, and initiating required notifications in accordance with 10 CFR 50.72 and procedure DB-OP-00002, "Operations Section Event/Incident Notifications and Actions". Additionally, the situation is assessed for impact on Technical Specification or other applicable system, channel, or train operability. If required, immediate actions are taken to mitigate the situation.

2. Perform Initial Assessment - The assigned responsible organization evaluates the problem, determines affected equipment and quality classification, determines if the issue is a condition adverse to quality, categorizes the condition as Category 1, 2, 3, or 4, assigns a weighting factor, and determines if Corrective Action to Prevent Recurrence (CATPR) is required.

   A Category 1 PCAQR is defined as a Significant Condition Adverse to Quality with escalated levels of management attention and involvement. A Category 2 PCAQR is defined as a Significant Condition Adverse to Quality. A Category 3 PCAQR is defined as a Condition Adverse to Quality. A Category 4 PCAQR is defined as a minor situation which does not warrant treatment as Category 1, 2, or 3 and that may be processed via other work/action programs.

For issues categorized as significant conditions adverse to quality, additional actions are required. Refer to the following section (D.3.0) titled "Significant Conditions Adverse to Quality".

3. Propose Remedial Action - The assigned responsible organization formulates a problem statement, determines apparent causes, evaluates operability, determines 10 CFR 21 reportability, proposes restoration actions, and identifies task assignments and responsible organizations for completion, including due dates. An experience review may be performed to determine if the issue has previously occurred. If appropriate, based on the significance of the issue and the extent of condition evaluated, the issue is reviewed by a supervisor/manager peer group, and the actions are tracked to completion. The peer group may also request corrective actions if not already provided.

4. Propose root cause/CATPR (if appropriate) - The assigned organization determines and documents the cause, proposes the preventive actions and justification, identifies tasks to be performed and determines due dates. For Significant Conditions Adverse to Quality, a root cause coordinator will coordinate the root cause investigation. Depending upon the category and weighting factor, the PCAQR may receive peer group review.

5. Complete Proposed Corrective Action - If necessary, the assigned organization completes actions and attaches appropriate documentation.

6. Verification - A close-out review is performed by the Quality Control organization. Assistance is also provided by the Quality Assessment organization for PCAQRs related to audits/surveillance findings, and by Regulatory Affairs for PCAQRs related to regulatory commitments. If appropriate for the significance of the PCAQR, this review includes an independent verification that the remedial actions and CATPR were performed.

The PCAQR process continues to be monitored through programs such as corrective action audits. In 1993, Audit Report AR-93-CORAC-02 and Management Corrective Action Report (MCAR) 93-003 were written to identify several opportunities for improvement with the PCAQR process. Areas such as problem investigation, corrective action, trending effectiveness, and PCAQR initiation threshold were identified for improvement.

To assess the inadequacies, a team was created which included individuals with varied levels of corrective action involvement from a cross-section of site organizations. Each of the identified areas was addressed, and actions were recommended and implemented to improve the process.

D.3.0 Significant Conditions Adverse to Quality

If a PCAQR issue is evaluated and determined to be a Significant Condition Adverse to Quality (Category 1 or 2), the respective Superintendent or Manager is notified of the heightened significance. Additionally, concurrence from the Director - Nuclear Assurance is obtained. This notifies the Director that a significant condition adverse to quality has been identified. The Vice President - Nuclear and each affected Director are also notified of the condition. The Manager - Nuclear Safety and Inspection coordinates the executive review of remedial actions and root cause/corrective action to prevent recurrence evaluations. The Manager - Nuclear Safety and Inspection also discusses the scope and status of Category 1 PCAQR corrective actions with the Vice President - Nuclear and the Company Nuclear Review Board.

D.4.0 Nuclear Assurance Audits

The DBNPS Quality Assessment (QA) Organization implements 10 CFR 50 Appendix B Criterion XVIII, USAR, and NQAM requirements through a series of scheduled audits performed to meet program and Technical Specification requirements. Additionally, USAR section 17.2.18, "Audits", defines the requirements for audits, including performance, reporting, and follow-up. Audit Finding Reports (AFRs) and PCAQRs are used by the QA organization to identify nonconformances or program deficiencies noted during the internal and external audit process. Audit Findings Reports are used to document program deficiencies or nonconformance details, identify any recommended corrective actions, document a root cause evaluation, and identify any remedial corrective actions to prevent recurrence. In addition, AFRs which may affect personnel or plant safety, operability, or may be reportable are reported immediately to the Manager - Quality Assessment and the Shift Supervisor and documented on a PCAQR. The auditor has responsibility for: concurrence with proposed corrective actions; verifying the effectiveness of the completed corrective actions through the close-out process; and performing follow-up effectiveness evaluations of the corrective action during subsequent audit activities.

D.5.0 Trending

The DBNPS Nuclear Quality Assurance Program utilizes statistical process control techniques to trend program and activity deficiencies as described in procedure NG-NA-00711, "Quality Trending". This trend program evaluates site corrective action documents (i.e., PCAQRs and audit findings) as well as external assessments such as Institute of Nuclear Power Operations (INPO) and Joint Utility Management Association (JUMA) findings. If this trend information exceeds an upper control limit, a Suspected Trend Investigation Report (STIR) is generated. The cognizant organization is notified of the suspected trend and an evaluation of this information is performed. The Quality Assessment Manager concurrence is required on STIR evaluations. If the information indicates an emerging trend over several months, the cognizant organization is notified and an investigation is requested. If either of these reviews confirms that an adverse quality

trend exists, a PCAQR is initiated to document the Significant Condition Adverse to Quality, and determine the root cause and corrective actions to prevent recurrence. These trend results are communicated to site management through the "Quality Trend Summary Report".

D.6.0 Self Assessments

Self-assessments are another vehicle that DBNPS uses to identify and correct program weaknesses. Self-assessments cover a wide spectrum of methods and practices of being self critical. These practices start with individual workers conducting critical self-evaluations of their work practices as well as initiation of PCAQRs when programmatic or human performance deficiencies are noted. Self-assessments also include a formalized process that utilizes an established self-assessment guideline ("Tools for A Successful Self-Sponsored Assessment") to document program reviews, potential improvements, and the initiation of corrective actions. In each case, responsible management concurs with the proposed corrective actions. Self-assessments are also incorporated into the day-to-day culture and operations and include the use of activity tracking logs to perform regularly scheduled assessments, such as control room monitoring and plant tours. Monitoring selected plant equipment through the use of engineering System Performance Books by the responsible System Engineer, supervisory oversight, and coaching in the maintenance areas provides added assurance that equipment and personnel work practices are meeting management expectations. These types of self-assessment activities promote problem identification and corrective action implementation.

D.7.0 Ombudsman Program

The DBNPS Ombudsman Program provides another method for plant personnel to confidentially report perceived nuclear safety or quality concerns. The Ombudsman Program is described in procedure NG-NA-00124, "Davis-Besse Ombudsman Program". The Ombudsman reports employee concerns to the Vice President - Nuclear. The program provides for addressing concerns at the appropriate management level necessary for resolution. The program is accessible by all personnel, including contractors and consultants. Concerns are received from individuals by direct contact with the Ombudsman or via telephone or mail. Concerns may be submitted anonymously. At the time of termination, resignation, or contract completion, individuals are given the opportunity for an exit interview with the Ombudsman or a supervisor. Concerns reported to the Ombudsman are handled with the highest possible degree of confidentiality for each situation.

The Ombudsman encourages individuals to resolve their concerns through normal plant management and quality assurance channels using PCAQRs, work requests, procedure change requests, or modification requests, as appropriate. However, the Ombudsman is available to all individuals who choose, for whatever reason, not to pursue the normal problem resolution channels. When a safety, quality, security, or radiological control concern is received, the Ombudsman works with the Vice President - Nuclear to determine the most appropriate plant or corporate organization to investigate the concern.

The investigating organization determines whether the problem should be documented in the corrective action process. The Ombudsman Program is separate from the corrective action process, but serves as an alternate entry point into the formal corrective action process. In all cases, any actions required for resolution of quality or safety problems are determined and tracked to completion. The individual who reports a concern is apprised of the outcome of the investigation, if the individual provided his or her name. The Ombudsman also reports the results of the investigation to the Vice President - Nuclear (prior to notifying the person who raised the concern in most cases) for concurrence that adequate evaluation of the situation was completed and proposed resolution is satisfactory. Periodic reports are given to the Vice President - Nuclear on the status of open/unresolved concerns.

NRC Request for Information:

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

Toledo Edison Response:

E.1.0 Course of Action Program

The current programs to capture, control and maintain the design and configuration of the DBNPS were largely initiated and developed as a result of the DBNPS Course of Action Program following the June 9, 1985, Loss of Feedwater Event. These programs and efforts were described in response to Items (b) and (c) and are briefly reiterated below.

E.1.1 System Review and Test Program

The DBNPS established the "System Review and Test Program" (SRTP) during the June 9, 1985 event outage to address concerns about the adequacy of safety systems and engineered safety features at the DBNPS. The SRTP conducted system level reviews to identify functions specified in the Operating License Technical Specifications (TS) and the Updated Safety Analysis Report (USAR).

Following these reviews, system restart tests were developed and performed to provide reasonable assurance that systems were capable of performing their intended functions.

The SRTP effort was reviewed by an NRC Staff and Staff contractor inspection team. The initial onsite coverage began during the formulation phase of the SRTP. This provided the opportunity for Staff concerns and comments to be incorporated into the program as it developed. As reported in NUREG-1177, "Safety Evaluation Report Related to The Restart of Davis-Besse Nuclear Power Station, Unit 1, Following The Event of June 9, 1985", the actions taken and testing completed "provided reasonable assurance that the plant has been improved beyond or restored to its original licensing basis..."

E.1.2 Configuration Management

As part of the DBNPS "Course of Action Program", the Configuration Management Program was established to document the design basis and configuration of the plant. The objectives of the configuration management program were to:

1) determine the as-built configuration of the plant by conducting system walkdowns to collect equipment information;

2) use the equipment data to validate vendor and Toledo Edison documentation;

3) use the equipment data to help develop system descriptions which are used for detailed design work and for the support of station activities;

4) establish a "living" data base; and

5) develop procedures to keep validated data base information current.

In meeting these configuration management objectives, the following products were developed: System Descriptions; Operational Schematics; Davis-Besse Configuration Equipment Summary (DBCES); and Design Criteria Manual (DCM).

E.1.3 Equipment Walkdowns

Various equipment walkdowns were conducted to document, verify, and update the design and configuration of the DBNPS. These included: walkdowns of equipment as part of the Equipment Qualification (EQ) Program; and Confirmatory Action Letter (CAL) 85-013 piping system support reinspections and piping system stress re-analysis.

E.1.4 Program Improvements

Along with the equipment walkdowns and system reviews and tests that were conducted as described above, program improvements were made to maintain and control the design, configuration and regulatory history that was established during these efforts. These program improvements are described as follows:

1) A commitment management program, the Toledo Edison Regulatory Management System (TERMS), was developed for managing commitments to the Nuclear Regulatory Commission (NRC), managing NRC requirements and maintaining a historical file of those commitments. The TERMS provides a data base to aid in managing commitments which require ongoing compliance, such as procedural actions and programs.

2) The processes used to change the facility were improved. Examples of improved processes are: the modification process; the procedure change program, including the Procedure Writers Guide and Validation Guide, the Qualified Reviewer Program, the 10 CFR 50.59 process, including training improvements, and maintenance and operations procedure improvements. These programs were described in detail in response to Item (a) and (b).

3) The Systems Engineering Section (now Plant Engineering) was also established in 1986. System Engineers are assigned to major plant systems and are responsible for the safe, reliable, and efficient performance of assigned systems. The system engineers are familiar with aspects of a particular system and provides a knowledgeable contact point for system problem resolution. The concept of "system ownership" promotes sound engineering input, resulting in effectively designed plant modifications. Plant Engineering is also responsible

for maintaining an appropriate preventive maintenance program for system components, developing troubleshooting plans, conducting problem investigations, and performing root cause determinations.

E.2.0 Internal Reviews

E.2.1 ISE Reviews

The DBNPS Independent Safety Engineering (ISE) Unit functions independently of other activities to review DBNPS engineering, design, operation, maintenance, plant modification, administration, and quality assurance activities. As described in response to Item (b), the ISE Unit has conducted several significant reviews of systems and plant programs. The reviews concluded that although weaknesses were identified, the safety systems were capable of performing their intended functions. The assessments also concluded that, generally, there were adequate programmatic controls in place in the design, maintenance, operation, testing, and training areas. Weaknesses identified during these assessments were tracked and appropriately dispositioned.

As discussed previously, the ISE reviews identified weaknesses in design calculation documentation. The DBNPS calculations are to be reviewed during the Design Basis Validation Program as described in the cover letter.

E.2.2 QA Audits and Surveillances

A review of Quality Assurance (QA) audits and surveillances conducted over the last six years indicates that when engineering programmatic deficiencies were identified (configuration and modification control activities) corrective actions were initiated and completed to resolve the identified concerns.

A review of audits of the corrective action process indicates that the current process is being used effectively to identify and resolve plant concerns. Past corrective action program weaknesses have been identified and were corrected through the Management Corrective Action Report (MCAR) process as described in Section D.2.0, Potential Conditions Adverse to Quality.

E.2.3 Conclusion

Based on a review of ISE reviews and QA audits, although deficiencies were identified, the safety systems were capable of performing their intended functions. When programmatic deficiencies were identified (configuration and modification control activities), corrective actions were initiated to resolve the identified concerns. The assessments also concluded that generally there were adequate programmatic controls in place in the design, maintenance, operation, testing, and training areas.

E.3.0 NRC Assessments

E.3.1 NRC System Functional Inspections

Various NRC assessments have been conducted in recent years such as: Electrical Distribution System Functional Inspection (EDSFI); Interfacing System LOCA Inspection (ISLOCA); and Service Water System Operational Performance Inspection (SWSOPI). These system-based SSFIs concluded that the system's design and operation were capable of performing the safety functions required by the design basis for the respective system.

A weakness identified in these inspections was in the area of configuration control. This weakness was described in NRC Inspection Reports 93-016 and 93-019, as addressed in the Toledo Edison letter dated March 14, 1994 (Serial Number 1-1036). Corrective actions included:

1) Detailed plant system walkdowns were conducted as discussed in response to Item (c). Management Corrective Action Report (MCAR) 93-002 was initiated to document configuration control deficiencies. These walkdowns did not identify any system or equipment operability issues.

2) A policy was issued and procedure changes made for configuration control of positionable components. The policy was developed by a multi-disciplined management team.

3) An evaluation and revision of the modification process was conducted to provide for affected document updating when plant configuration changes are made.

These corrective actions served to: 1) establish an accurate baseline for plant configuration; 2) provide assurance that future plant configuration changes are accurately and consistently reflected in plant documents; and 3) provide a detailed, well-communicated configuration control policy.

E.3.2 Program Inspections

Special NRC assessments were also conducted in the areas of: Maintenance; Design Changes and Modifications; and Engineering and Technical Support. The inspections concluded that the modifications reviewed were in accordance with programmatic and regulatory requirements.

The inspections of the DBNPS design changes and modifications identified a weakness in the 10 CFR 50.59 safety review/evaluation process as documented in Inspection Report 94-016, and as addressed in the Toledo Edison letter dated March 10, 1995 (Serial Number 1-1063).

Toledo Edison concluded that although the existing safety evaluation program was adequate to meet regulatory requirements and industry guidance, additional improvements should be made. Improvements included implementation of a new computer-based training program, revision of the 10 CFR 50.59 training manual, and revision of the safety review procedure and the

associated safety review screening form. The procedure changes require personnel conducting the safety reviews to provide a more extensive written justification for each answer on the screening form.

E.3.3 NRC Resident Inspections

A violation was issued in NRC Inspection Report 96-005 concerning the failure to perform a 10 CFR 50.59 safety evaluation for a change to the plant as described in the USAR. On December 15, 1995, the Primary Water Storage Tank (PWST), a tank described in the USAR as providing a water source to the plant, was drained and made unavailable for station use. This was completed without performing a safety evaluation to determine whether the change constituted a change to Technical Specifications or whether there was an unreviewed safety question.

As reported in Toledo Edison's letter dated December 19, 1996 (Serial Number 1-1112), the individuals involved used inappropriate judgment and considered it sufficient to permit draining of the tank under an MWO because it would provide for proper tag-out, removal from service, and restoration, if needed. The cause of this violation was poor judgment in implementation of site processes by the individuals involved. A safety evaluation was subsequently completed documenting that the abandonment of the PWST did not constitute a change to the Technical Specifications and did not represent an unreviewed safety question.

E.3.4 1996 Engineering Inspection

The most recent inspection of the DBNPS engineering and technical support functions occurred on July 29, 1996, through August 27, 1996, and is documented in Inspection Report 96-007. This inspection concluded that:

1) Plant (system) engineers were knowledgeable of their systems and actively involved in ensuring the reliability of the system for which they had responsibility. Plant engineers were knowledgeable of the various regulatory requirements and programs which affected their performance.

2) Facility modifications were implemented in accordance with the applicable installation and testing requirements.

3) Response to industry events were usually thorough and demonstrated a good questioning attitude on the part of plant engineering.

E.3.5 Conclusion

These NRC inspections support the DBNPS internal assessment findings. Individual violations have been identified; however, the safety systems were capable of performing their intended functions. When engineering programmatic deficiencies were identified (e.g., configuration and modification control, and 10 CFR 50.59 evaluation activities), appropriate corrective actions were initiated to resolve the identified concerns and are tracked to completion via the DBNPS commitment management system (TERMS).

E.4.0 Recent Reviews and Improvements

Several recent improvements have been made as a result of a variety of PCAQR identified weaknesses, self-assessments, industry experience reviews, LERs, and DBNPS-initiated programs. Significant reviews and improvements are summarized below.

E.4.1 Industry Events Review

A weakness was recently identified in the DBNPS industry events assessment program as discussed in Licensee Event Reports (LER) 96-002, "Potential Loss of Remote Shutdown Capability due to MOV Fire Induced Damage", dated July 31, 1996, and LER 96-004, "Inadequate Compensatory Actions for Thermo-Lag for Radiant Energy Shields", dated May 17, 1996.

In addition to the corrective actions for the specific conditions reported in the LERs, the DBNPS process for review of NRC Information Notices was modified to include Information Notice evaluation through the DBNPS PCAQR corrective action process. The PCAQR process provides operability and reportability reviews; a more structured timeline for issue review and resolution; prompt significance reviews; multi-discipline review, if deemed appropriate based on significance of this issue; and independent review at the close out of the PCAQR.

E.4.2 24-Month Fuel Cycle Reviews

The conversion of the DBNPS from a 18 month fuel cycle to a 24-month fuel cycle requires reviews and reanalyses, as necessary. Issues requiring review include: USAR Chapter 15 accident analysis, Technical Specification-required instrument setpoint drift studies, and environmental qualification of equipment. These reviews are currently being performed.

During this effort, the basis for extension of Surveillance Requirement 4.5.2.b for venting of Emergency Core Cooling System (ECCS) HPI piping was reviewed. It was discovered that the high point of the discharge piping downstream of the normally closed valve HP2A had no manual vent valve, and that the surveillance procedure DB-SP-03212, "Venting of ECCS Piping", did not specify a venting method for that line. This was reported in Licensee Event Report (LER) 96-001, dated April 3, 1996.

The apparent cause of the occurrence was an inadequate safety review/evaluation for the modification which changed the configuration of Make-up and High Pressure Injection in 1989. As reported in LER 96-001, improvements have been made in the safety review and evaluation process since implementation of this plant modification. Recent improvements in the process were made in response to NRC Inspection Report 94-016 and are described above.

E.4.3 Compliance Emphasis

On March 28, 1996, it was determined that the testing procedures associated with the testing of charcoal adsorbers located in the Hydrogen Purge System (HPS), the Emergency Ventilation System (EVS), and the Control Room

Emergency Ventilation System (CREVS) were not in strict compliance with the Surveillance Requirement for testing of the charcoal. There was no question of the ability of the charcoal adsorbers to perform their safety function; however, the systems were declared inoperable due to the inability to literally comply with the Surveillance Requirement. This was reported in LER 96-003, dated April 29, 1996.

Licensee Event Reports 96-001 and 96-003 heightened the sensitivity of station personnel to compliance issues. To emphasize the importance of strict compliance with regulatory requirements, the USAR and procedures, training was given to Senior Reactor Operators, Plant Engineering, Design Basis Engineering, the Station Review Board, the Company Nuclear Review Board, and Independent Safety Engineering.

E.4.4 USAR Review

In March 1996, the NRC issued Information Notice 96-17, "Reactor Operation Inconsistent With The Updated Final Safety Analysis Report". This Notice described issues that were discovered at the Millstone Unit 1 facility, where the utility's practices and actions did not match that described in their USAR.

This Information Notice, combined with the lessons learned from LERs 96-001 and 96-003, formed the basis for the Cycle 10/10 Refueling Outage (RFO) review of the DBNPS Updated Safety Analysis Report (USAR). During this review by the "cognizant units", specific attention was given to the review of operational and design aspects of systems and components described in the USAR, ensuring that the text, tables, and figures of the USAR accurately depicted the facility and its operation.

The Cycle 10/10 RFO USAR update was submitted on December 2, 1996. While the majority of the items have been resolved and incorporated into Revision 20 to the USAR, several items were not resolved in time for incorporation into this update. As these items are resolved, any resulting USAR changes will be processed and "posted" and therefore, will be available for personnel performing 10 CFR 50.59 safety review/evaluations.

Although the review did identify discrepancies between the USAR and the as-built plant and operating procedures, no unreviewed safety questions were identified. A root cause evaluation for the high number of discrepancies identified during this USAR review effort is being performed under the PCAQR reporting process.

The Cycle 10/10 RFO USAR review identified one condition that was subsequently reported in LER 96-007. As reported in LER 96-007, a review of the calculations for USAR Figure 9.4-13, "Relationship of Measured Pressure vs. Leakage Area in the Control Room", revealed that the total CREVS flowrate of 3,300 cubic feet per minute (cfm) was erroneously used to evaluate the leakage through the Control Room boundary. The correct value is 300 cfm which is the amount of make-up air flow that is brought into the control room from outside air during the pressurization mode. The apparent cause was an error in the original system design calculation, which was performed in 1973. Assumptions used in this analysis are being re-evaluated and the calculations, USAR, and affected procedures will be revised

accordingly. Other original design basis assumptions and calculations are being reviewed as part of the DBNPS Design Basis Validation Program as described in the cover letter.

E.4.5 Licensee Event Report (LER) Review

The DBNPS LERs written in the last three years were reviewed for issues reported in accordance with 10 CFR 50.73(a)(2)(ii)(b) as being outside the design basis. This review was performed to determine if a trend existed in reportable events in the areas of: configuration control; or system design; or configuration outside of the design basis. In addition to the LERs described above in Section E.4, the review results are as follows:

1) LER 95-001, dated January 26, 1995, reported a condition that was outside the design basis. The Loss of Coolant Accident (LOCA) analysis determines the allowable linear heat rate limits contained in the DBNPS "Core Operating Limits Report" (COLR). The fuel vendor determined that the LOCA analysis results for the 2-ft core elevation were potentially non-conservative for the Cycle 10 core. The DBNPS Technical Specifications operating limits were unaffected, therefore, this condition had minimal safety significance. However, the allowable linear heat rate contained in the COLR was affected under certain core flux imbalance or tilt conditions (e.g., dropped rod). The reported condition resulted from errors contained in the fuel vendor's computer code and modeling methodology for the approved 10 CFR 50, Appendix K LOCA Evaluation Model.

2) One LER was reported involving a procedural use error that resulted in the facility being in a condition outside the design basis. LER 96-005, dated August 16, 1996, reported lifting the Reactor Vessel Head Lifting Tripod (RVHLT) and improperly traversing a portion of the open reactor vessel with fuel in the reactor. The RVHLT is considered a heavy load and is procedurally restricted from movement over the open reactor vessel with irradiated fuel in the reactor. The cause of this LER was a failure to follow procedures.

3) One LER was reported in the last three years involving maintenance of the facility that resulted in the system being returned to service in a condition outside its design basis. Licensee Event Report 96-006, dated June 13, 1996, reported the 1.5 inch lip around the top of Reactor Coolant Pump Motor (RCPM) 2-1 was not in place after RCPM replacement in 1993 as required by 10 CFR 50, Appendix R analysis. The appropriate work planning procedures are being revised to provide proper verification of RCPM oil collection system equipment during future RCPM work. This condition was discovered during an engineering walkdown.

4) LER 96-010, dated January 13, 1997, reported a condition that was potentially outside the design basis of the plant. Both station vent radiation monitors had been removed from service and it was determined that both trains of the Control Room Emergency Ventilation Service

(CREVS) may not have been capable of performing their intended safety function of isolating the control room during an 87-minute period on December 10, 1996. This condition occurred when both radiation monitors were removed from service for maintenance and filter replacement, respectively. Neither procedure for these activities directed the operators to take actions due to having both station vent radiation monitors out of service. The Surveillance Requirements for the CREVS and the Emergency Ventilation System (EVS) were reviewed for additional similar requirements. No other similar requirements were found.

E.4.6 10 CFR 50.72 Notifications

In addition to the LER review, there are two conditions that have been recently reported in accordance with the requirements of 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors", as being potentially outside the design basis of the plant.

1) On February 3, 1997, it was discovered that instrument tubing associated with each Reactor Coolant Pump Motor (RCPM) oil lift system was outside of the enclosure designed to contain oil spray from potential leak sites as required by 10 CFR 50 Appendix R. This protection is necessary when the oil lift pumps are in operation and this portion of the system is pressurized. The lift pumps are normally operated during initial RCPM startup.

2) On January 22, 1997, it was determined that portions of piping associated with certain containment penetrations could exceed the ASME Section XI interim code allowable values. This concern was identified as a result of the review being conducted in response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions". Engineering evaluations have concluded that the penetrations of concern are capable of maintaining containment integrity during a large break Loss of Coolant Accident (LOCA).

These conditions are currently being evaluated by the DBNPS PCAQR process for apparent cause and corrective actions, including those actions needed to reduce the probability of similar events occurring in the future. These results will be reported as required by 10 CFR 50.73, "Licensee Event Report System". Additionally, as required by 10 CFR 50.73, the individual LERs will contain an analysis of past LERs to determine if the occurrence is similar to previous events.

E.4.7 Conclusion

Although conditions were identified as being outside the design basis, the analysis of the individual LERs concluded that mitigating factors were in place to reduce the consequences of the postulated events for each condition reported. Review of the causal factors of each LER, indicates that similar conditions were not evident and these conditions are not indicative of programmatic weaknesses related to: design basis information being translated into procedures, configuration control, or plant systems being adequately designed and tested.

Furthermore, identification, reporting, and resolution of these conditions indicates that site programs and personnel are generally effective in identifying potential conditions adverse to quality. When potentially adverse conditions are identified, operability determinations are made, root causes are determined and corrective actions are implemented. Several of these conditions were, in part, identified due to the effectiveness of the corrective action process in heightening site awareness of the importance of strict compliance with design and licensing basis requirements.

E.5.0 Ongoing Assessments

The DBNPS performs oversight of the design basis control and implementation process through a variety of internal and external assessment processes, including QA audits and surveillances, individual department self assessments, and trends identified through procedure NG-NA-00711, "Quality Trending". When these assessments uncover program weaknesses, corrective action is initiated to resolve the concern. These efforts have improved, and will continue to improve the design and configuration management processes.

The Quality Assessment Section audits design basis information (modification control, configuration control, etc.) during their 10 CFR 50, Appendix B biennial audit of design engineering activities. These audits are also supplemented by periodic surveillance activities that evaluate individual engineering functions such as the drawing change process, the setpoint change process, the equipment qualification program, etc. Program weaknesses identified during these activities are documented on audit finding reports or PCAQRs, and corrective action is verified by the QA organization.

E.6.0 Overall Conclusion

Toledo Edison believes that the programs and assessments described in response to Item (b), (c), and (e) provide reasonable assurance that:

> The safety systems are capable of performing their intended functions. The programs in place at the DBNPS, as described in the response to Item (a), are adequate to control, evaluate, change and maintain system design information, regulatory requirements, and system configuration. The systems perform and are tested appropriately with respect to the design basis.
>
> Systems, structures, and components are configured consistent with the design bases and are maintained by adequate programs. The documents and data bases which include configuration information or provide direction related to plant operation are maintained consistent with the as-built plant design basis is in accordance with the DBNPS Configuration and Design Control processes.

Systems, structures, and components are tested and will perform as required to support the design bases. Testing procedures and programs are maintained consistent with the as-built plant design basis is in accordance with the Configuration and Design Control processes.

Various reviews identified the need for improvement in some design calculation documentation. As previously discussed in the cover letter, the DBNPS is undertaking a Design Basis Validation Program to validate the existing system design bases, with emphasis in reviewing the design basis analyses and supporting calculations.

The DBNPS recognizes the importance of maintaining conformance between the USAR, Technical Specifications, plant drawings, plant procedures, and the as-built facility, commitments, and its design basis. Ongoing internal, external and audit assessments of these controls provide the opportunity to identify, correct, and continually improve the programs and processes.

References

DBNPS References

- DBNPS Updated Safety Analysis Report (USAR) Chapter 17.2, Quality Assurance During the Operations Phase.
- Davis-Besse Nuclear Power Station (DBNPS) Nuclear Quality Assurance Manual. (NQAM)
- DBNPS ASME Quality Assurance Manual. (AQAM)
- DBNPS Nuclear Group Procedure NG-EN-00307, Configuration Management.
- DBNPS Nuclear Group Procedure NG-EN-00301, Plant Modifications.
- DBNPS USAR Chapter 15, Accident Analysis.
- DBNPS Nuclear Group Procedure NG-EN-00310, Setpoint Control.
- DBNPS Nuclear Group Procedure NG-DB-00205, Plant Maintenance.
- DBNPS Nuclear Group Procedure NG-EN-00313, Control of Temporary Modifications.
- DBNPS Procedure DB-DP-00007, Control of Work.
- DBNPS Nuclear Group Procedure NG-NS-00802, Commitment Management.
- DBNPS Nuclear Group Procedure NG-NA-00115, Control of Procedures.
- DBNPS Procedure Writers Guidelines.
- DBNPS Procedure Validator's Guidelines.
- DBNPS Nuclear Group Procedure NG-EN-00304, Safety Review and Evaluation.
- DBNPS Nuclear Group Procedure NG-NS-00801, Operating License Amendments.
- DBNPS Nuclear Group Procedure NG-VP-00132, Qualified Reviewer Program.
- DBNPS Nuclear Group Procedure NG-NS-00806, Preparation and Control of USAR Changes.
- DBNPS Fire Hazards Analysis Report.
- DBNPS Technical Requirements Manual.
- DBNPS Course of Action.

- DBNPS Design Criteria Manual.
- DBNPS System Descriptions.
- DBNPS System Review and Test Program. (SRTP)
- DBNPS Procedure EN-DP-01150, System Descriptions.
- DBNPS Nuclear Group Procedure NG-DB-00116, Outage Nuclear Control.
- DBNPS Procedure DB-OP-06904, Shutdown Operation.
- DBNPS Procedure DB-OP-00015, Safety Tagging.
- DBNPS Procedure DB-OP-00016, Removal and Restoration of Station Equipment.
- DBNPS Procedure DB-OP-00307, Station Configuration Control.
- DBNPS Procedure DB-OP-04004, Locked Valve Verification.
- DBNPS Procedure DB-OP-04005, Capped Valve Verification.
- DBNPS Procedure DB-OP-00005, Operator Logs and Rounds.
- DBNPS Procedure DB-OP-00000, Conduct of Operations.
- DBNPS Nuclear Group Procedure NG-DB-00202, Test Control.
- DBNPS Procedure DB-OP-00307, Station Configuration Control.
- DBNPS Nuclear Group Procedure NG-NA-000702, Potential Condition Adverse to Quality.
- DBNPS Nuclear Group Procedure NG-NA-00701, Nuclear Assurance Audit and Surveillance.
- DBNPS Nuclear Group Procedure NG-NA-00711, Quality Trending.
- DBNPS Tools for a Successful Self-Sponsored Assessment
- DBNPS Nuclear Group Procedure NG-NA-00124, Davis-Besse Ombudsman Program
- DBNPS USAR Table 17.2-1, Applicable NRC Regulatory Guides, ANSI Standards, and Industry Codes
- DB-OP-00002, Operations Section Event/Incident Notifications and Actions
- DB-OP-00008, Operation and Control of Locked Valves

- DB-OP-00009, Operations and Control of Capped Valves
- DB-OP-06911, Prestart-up Checklist
- DB-PF-00002, Preventive Maintenance
- DB-PF-01025, Maintenance Testing Requirements
- DB-SP-03212, Venting of ECC Piping
- EN-DP-01072, Modification Test Requirements
- EN-DP-01074, Predictive Maintenance
- EN-DP-01301, CEMAP
- NG-EN-00314, Inservice Inspection
- NG-NA-00701, Nuclear Assurance Audit and Surveillance
- NG-NA-00702, Potential Condition Adverse to Quality Reporting
- NG-NA-00711, Quality Trending

Other References

- American National Standards Institute (ANSI) N45.2.11 - 1974, Quality Assurance Requirements for the Design of Nuclear Power Plants.
- ANSI/American Nuclear Society (ANS) 3.2 - 1982, Administrative Controls and Quality Assurance for Nuclear Power Plants.
- Electric Power Research Institute (EPRI) TR-103586, Guidelines for Optimizing the Engineering Change Process for Nuclear Power Plants.
- ANSI N18.1 - 1971, Selection and Training of Nuclear Power Plant Personnel.
- NUMARC 90-12, Design Basis Program Guidelines.