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Ladies\Gentlemen:

DOCKETS 50-266 AND 50-301
RESPONSE TO REQUEST FOR INFORMATION PURSUANT TO
10 CFR 50.54(f) REGARDING ADEQUACY AND AVAILABILITY OF DESIGN BASES
INFORMATION, POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

This letter responds to your letter dated October 9, 1996, requesting Wisconsin Electric (WE) to provide information to the NRC staff, pursuant to 10 CFR 50.54(f) regarding the adequacy and availability of design bases information for the Point Beach Nuclear Plant (PBNP).

This response is structured to provide a description of the PBNP Design Basis Document (DBD) program followed by specific responses (as attachments to this letter) to the five specific requests for information in your letter:

(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 (Attachment A);

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

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

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

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

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In responding to these items above, various processes, programs, and procedures related to PBNP design control, configuration control, and corrective actions are identified and described. The description of such processes, programs, and procedures in this letter is not intended to create new commitments (unless explicitly stated otherwise) and is not intended to preclude changes in accordance with existing normal administrative practices.

Focus of Design Bases

We understand, both from the October 9 letter requesting information and from industry communications, that the NRC's interest focuses on "design bases" -- how we know the PBNP design bases, how we maintain plant configuration consistent with design bases, and how we identify and correct any discrepancies. We have endeavored to address these issues in this response recognizing the NRC definition of design bases from 10 CFR 50.2, as described and explained in footnote 4 of the October 9 letter, which states:

> "Design bases mean that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design..."

This footnote also goes on to state that:

> "The design bases of a facility, as so defined, is a subset of the licensing bases and is contained in the Final Safety Analysis Report (FSAR). Information developed to implement the design bases is contained in other documents, some of which are docketed and some of which are retained by the licensee."

Based upon our experience in researching and compiling design basis information, Wisconsin Electric has applied the broader defined term "Engineering Design Bases" to the PBNP Design Basis Document (DBD) program. Engineering Design Bases are defined in NUREG-1397 as:

> "the entire set of design constraints that are implemented, including those that are (1) part of the current licensing basis and form the bases for the NRC staff's safety judgment and (2) those that are not included in the current licensing basis but are implemented to achieve certain economies of operation, maintenance, procurement, installation, or construction."

The objective of preparing DBDs for PBNP is to capture the functions and parameters that describe a system or topical area design bases. As a result, both safety-related and non-safety-related functions and associated parameters are described in DBDs. In addition, our DBDs include, wherever possible, supporting design information which provides the reason why a particular design basis exists, thereby helping to establish an understanding of the design bases. A PBNP DBD contains the engineering design bases for a system or topical area. Therefore, engineering design basis information which is captured by PBNP DBDs encompasses the 10 CFR 50.2 design bases.

Design Basis Document (DBD) Program Description

The purpose of the PBNP DBD program is to prepare DBDs to provide centralized access to design basis information and supporting design information. As defined in the PBNP DBD Program Manual, a DBD is "*the document or collection of documents which contains the design basis for a component, structure, system, or topic, and which references and/or contains any supporting information necessary to supplement the design basis.*" The DBD contains the engineering design bases information as described in the above paragraphs for a system or topical area.

Wisconsin Electric recognized in the late 1980s, that for various reasons, there was a need to retrieve original PBNP design bases and to compile the design basis information. As a result, WE began the development of the PBNP DBD Program Plan in 1989. The PBNP DBD program was developed by WE personnel and contractors both experienced in writing DBDs for other utilities. The guidance provided in NUMARC 90-12, "Design Basis Program Guidelines," was also used in the development of the DBD program, specifically with respect to the DBD program scope, format and content, validation, and management of discrepancies. We believe our program is consistent with the guidelines of NUMARC 90-12 and considers the guidance of NUREG-1397, which describes the NRC assessment of several utility design document reconstitution programs. Formal preparation of DBDs for PBNP began in 1991 and has steadily continued since that time. We have maintained as a priority the retrieval and compilation of PBNP design basis information, recognizing the NRC expectations in the 1992 NRC Policy Statement on "Availability and Adequacy of Design Bases Information at Nuclear Power Plants". All DBDs currently in the scope of this program are planned for completion and initial issuance by 1999.

An Administrative Manual Policy, AM 3-10 "Design Basis Documents" describes the Nuclear Power Business Unit (NPBU) policy for the conduct of the DBD program and describes how the DBDs may be used. The DBDs are provided for use in the preparation and review of modifications, performance of safety evaluations and engineering analyses, research of regulatory commitments, identification of test requirements, and understanding design requirements. DBDs may also be used to support operability determinations, respond to audit issues, and evaluate proposed technical specification changes. The preparation of DBDs and the administration of the DBD program are governed by the PBNP DBD Program Manual. The DBD Group in NPBU has the responsibility and ownership for preparation and maintenance of the PBNP DBDs, and for assisting with response to design basis issues that arise.

Scope

The scope of the DBD program currently includes 41 DBDs (29 system DBDs and 12 topical area DBDs) and one DBD position paper. A list of these DBDs is contained in Attachment F. This attachment also indicates which DBDs are completed, in progress, or not yet started, and provides the actual or expected completion date by year. The scope and schedule for the PBNP DBD Program reflects the status of a voluntary program initiated by WE. It is our intent to fully complete this program in a timely manner, in accordance with the scope and schedule described in Attachment F. However, there may be a need in the future to alter the scope and/or schedule to appropriately address PBNP design bases needs.

The present DBD program scope is based on the following selection criteria:

- Safety-related systems
- Topical areas related to nuclear safety
- Topical areas related to augmented quality[1]
- Other systems with known or anticipated design issues
- Other systems / topics for which knowing the design basis will provide a long-term benefit

A comparison of the DBD program scope to Maintenance Rule risk-significant systems is provided in Attachment F. All Maintenance Rule risk-significant systems are included in the DBD program scope with the exception of the mechanical aspects of the Gas Turbine and its support systems.

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[1] Augmented quality items are non-safety-related items for which Wisconsin Electric has made a regulatory or design basis commitment; or, for plant availability reasons, Wisconsin Electric has implemented special controls to assure reliability.

Design Document Retrieval

To support the PBNP DBD development effort, two extensive design document retrieval efforts were undertaken by WE: one with Westinghouse, the NSSS vendor; and one with Bechtel, the original Architect/Engineer. Wisconsin Electric led a Westinghouse Owners Group Design Document Program (DDP) Subgroup effort to retrieve, index, and optically store Westinghouse design information for the DDP Subgroup members. The types of documents retrieved as part of this effort included calculations, functional requirements, system standards, correspondence, specifications, Westinghouse Commercial Atomic Power (WCAP) reports, and safety evaluations. Over 9000 documents were made available by Westinghouse to WE as a result, with a detailed index and document images stored on CD-ROM. In addition, WE worked with Bechtel to obtain copies of historical microfilm records and hard copies of calculations which support the PBNP design. Nearly 200 microfilm cartridges and approximately 65 volumes of mechanical and civil hard-copy calculations were obtained in this effort. Review of these documents to retrieve design basis information is performed as part of the DBD preparation process, as appropriate.

DBD Preparation

The PBNP DBD Program Manual contains a detailed "Writer's Guide" (DBDP 4-1) which establishes guidelines for the content and format of system and topical area DBDs. It also includes procedures for DBD review and approval, validation, open item management, and DBD revision and maintenance. This manual is a controlled document, and is reviewed and revised as necessary when enhancements to the DBD program are made.

DBDs for systems and topical areas (such as Equipment Qualification or Post-Accident Monitoring) are prepared in accordance with the PBNP DBD Writer's Guide. The majority of the DBDs are written by experienced WE DBD group personnel in order to develop and retain PBNP design basis expertise within the NPBU. Vendors or contractors are used on a limited basis. WE has utilized Westinghouse to prepare DBDs reflecting original design basis information (pre-Operating License) for several NSSS systems, and to prepare Accident Analysis Basis Documents, which describe the bases for the FSAR Chapter 14 accident analyses.[2] Contractors are used for preparation of selected DBDs where specific external expertise on those systems or topical areas is desirable.

DBD Content

The content of each PBNP DBD is extensive, and describes system, structure, and component (SSC) or topical area design bases, as well as supporting design information. The content of each system DBD includes the following:

- System description and boundaries
- System design bases
    - System functional requirements
    - System performance requirements
    - Other design requirements
- Component design bases
    - Component functional requirements
    - Component performance requirements
    - Other design requirements
- Applicable codes, standards, and regulatory documents and how they apply to the SSC design bases
- Precautions and limitations for SSC operation that pertain to design basis requirements

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[2] The Accident Analysis Basis Documents were prepared with a different format from the other DBDs, which was more appropriate for the nature of the information being presented.

- SSC inspection and testing requirements related to design basis requirements
- SSC modification summaries
- SSC calculation summaries
- DBD reference list
- DBD open items

The content of each topical area DBD includes the following:

- Topical area description, history, and boundaries
- Topical area design bases
- Applicable codes, standards, and regulatory documents and how they apply to the topical area design bases
- Description of programs to implement and maintain the topical area design bases
- Topical area modification summaries
- Topical area calculation summaries
- DBD reference list
- DBD open items

DBD Review

The DBD review process is described in DBDP 4-4, "Design Basis Document Review and Approval". The process requires that draft DBDs, with their associated open items, receive a comprehensive review by a DBD engineer independent of the DBD author. They also receive technical reviews by cognizant engineering, operations, and quality assurance personnel prior to issuance. These reviews are intended to confirm that the design basis information has been correctly extracted from the source documents and that the DBD is complete and consistent with the design basis information. Guidelines for the reviewers are specified in DBDP 4-4.

Validation

All DBDs are validated (with limited, justified exceptions[3]) prior to being issued as Revision "0" DBDs. Attachment F lists the validation status of each DBD. The validation process is described in DBDP 4-5, "Validation of Design Basis Attributes". The purpose of the validation is to provide reasonable assurance that selected design basis attributes are properly and consistently implemented in the physical plant and in those documents important for the support of plant operation. The validation is a one-time effort.

For a system DBD, the validation is intended to provide reasonable assurance that:

- The system design has been implemented such that the system can accomplish its functions and meet its performance requirements.
- The system is adequately tested to demonstrate that it will accomplish its functions within its performance requirement limits.
- The system is properly operated during normal and accident conditions consistent with its design bases.

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[3] Exceptions to validating DBDs have been made for (1) the Chemical Volume Control System (CVCS) DBD when a WE Vertical Slice Audit had been recently performed and was confirmed to have covered the expected scope of the DBD validation, and for (2) Westinghouse-prepared Accident Analysis Basis Documents (AABDs), which are detailed summaries of the FSAR Chapter 14 accident analysis. The applicable information from these AABDs should be incorporated in the appropriate system or topical area DBDs, which are validated. The DBD position paper on Electrical and Mechanical Separation was not validated because it contained primarily guidance information, not design bases.

Similarly, for a topical area DBD, the validation is intended to provide reasonable assurance that the plant is configured, tested, and operated consistent with the topical area design bases.

The validation of each PBNP DBD is performed using a validation checklist, prepared by the DBD author and receiving an independent review, which documents the design basis attributes to be validated. The DBD validation procedure provides guidelines for selecting the design basis attributes to be validated. The validation is generally performed at the plant site, by a team independent of the DBD author(s), who utilize a combination of document reviews and plant walkdowns to obtain the information necessary to assess each attribute. Guidelines for the plant walkdowns are also included in the DBD validation procedure. Results of the validation, including detailed explanations for the conclusions for each attribute, and walkdown results, are documented in a detailed validation report. As appropriate, the validation results are included when finalizing the DBD as Revision "0".

Open Items

DBD open items may be identified as a result of DBD research, preparation, review, validation, or by DBD users. The identification and management of DBD open items is governed by DBDP 4-3, "Design Basis Open Item Management", and procedure NP 7.7.3, "As-Built Drawing Program and Design Basis Document Program Open Items". In accordance with these procedures, whenever a DBD open item is identified, it is documented and assessed to determine if a Condition Report is required. The Condition Reporting process, described in procedure NP 5.3.1 "Condition Reporting System", documents conditions adverse to quality and is used to initiate operability and reportability screenings. If it is determined that a Condition Report should be generated, then procedure NP 5.3.1 takes precedence over the DBD open item management procedures.

If a Condition Report is not judged to be necessary, the DBD open item is entered into and formally tracked in the Nuclear Tracking System (NUTRK). A recent change to the DBD open item process, which is currently being implemented, requires that an additional review of all DBD open items be performed by the DBD group with a licensed Senior Reactor Operator (SRO) and the responsible System Engineer. The purpose of this additional review will be to provide additional assurance that any safety, operability, or reportability implications associated with the open item are identified prior to DBD issuance.

The work priority and due date for resolving a DBD open item are assigned using NUTRK. The responsibility for resolving the open item may be assigned to a DBD engineer, or to other NPBU personnel, as appropriate. Open item resolution may require additional research, design document reconstitution, or initiation of additional technical activities. Again, if at any time during the open item research, review, or resolution process, it is determined that a Condition Report should be generated, then procedure NP 5.3.1 takes precedence over the DBD open item management procedures so that a Condition Report is prepared promptly. Once a DBD open item has been resolved, its closure is noted in the DBD and documented in NUTRK.

In addition to the DBD open item reviews described above, DBD open items are reviewed on a semiannual basis, at a minimum, to confirm that their status, method of disposition, work priority, and due date are appropriate for the nature of the item, with consideration given to the safety significance of the item and to the importance of the design basis information.

A special review of DBD open items was performed in December 1996 in response to questions raised by NRC Inspectors during an Operational Safety Team Inspection. This review utilized a lower threshold for Condition Reports than what had historically been used. All open items for DBDs which have been issued were re-reviewed for safety significance and operability / reportability implications by the DBD group with licensed SROs and System Engineering personnel. Condition Reports and prompt operability determinations were prepared as deemed

appropriate. Follow-up actions for the Condition Reports which were generated were documented and are being tracked in NUTRK in accordance with the Condition Report process. As a result of this special review, no equipment was determined to be inoperable, however, one four hour report was made to the NRC.

DBD Revision and Maintenance

Procedure DBDP 4-6, "Design Basis Document Revision and Maintenance", governs the process for DBD revision and maintenance. Each DBD is reviewed for revision within a maximum time interval of three years from the information cut-off date of the previous DBD revision. If significant changes to SSC or topical area design bases occur, DBDs may be revised more frequently in order to help keep current design basis information readily available to users. A revision to a DBD may be necessary to include the effects of licensing changes, modifications, evaluations, calculations, closure of DBD open items, and enhancements or clarifications to DBDs identified by DBD users. When each DBD revision is prepared, design basis information sources which are generated or changed since the information cut-off date of the previous revision are reviewed. DBD open items are reviewed and researched for additional information or possible closure. Prior to issuance, the DBD revision receives a review from an independent DBD engineer, and also a technical review as appropriate for the nature of the DBD changes.

Design Basis Accessibility

To make the design basis information accessible to all NPBU users, controlled copies of completed DBDs are located in multiple locations in our Milwaukee office and at PBNP. In addition, completed DBDs will soon be electronically accessible and word-searchable in a new PBNP electronic document management system which is currently being installed. Copies of DBD reference books, which are a compilation of the DBD reference documents that are not readily available, are located in the DBD group files in the Milwaukee office and in the main document control center at PBNP.

Adequacy and Availability of PBNP Design Bases Information

Wisconsin Electric believes that proper implementation of the PBNP DBD program described above, together with the programs and processes described in the responses to information items (a) through (d), provide reasonable assurance that PBNP is operated, maintained and configured consistent with its design bases as described in Attachments A through E.

Historically, Wisconsin Electric's internal measures of effectiveness have shown that the plant configuration and operation are generally consistent with the design bases. A review of the trends of assessments, audits and nonconformances would suggest that the number and significance of issues have remained relatively constant. When the results of these efforts are closely analyzed, however, it can be seen that programmatic problems have become fewer and the technical findings have become more sophisticated and complex. These results suggest that the processes and programs have improved and personnel are challenging calculations and assumptions. Overall, we have become more sensitive to design bases issues due to our DBD efforts coupled with the results of WE assessments and NRC inspection activities.

We therefore conclude that our processes and programs continue to become more effective and our threshold for identifying design related nonconformances has decreased. Wisconsin Electric, however, recognizes that additional program improvements can be made with regard to the consistency in implementation of PBNP design basis information and we are working to improve these areas. Areas requiring improvement have been identified over the past year by WE assessments and NRC inspections, with regard to PBNP procedures, work and test activities, licensing and design bases adherence, and the corrective action program.

Recently identified instances where procedures were found to be deficient with respect to design basis information may indicate weaknesses in maintaining procedures consistent with design basis information. As discussed in Attachment B, several recent initiatives have been completed and others are ongoing in an effort to strengthen the link between design basis requirements and the operating, maintenance, and testing procedures.

In addition, a preliminary NPBU Configuration Management (CM) Program Plan has been developed, and a group has been added to the organizational structure of NPBU whose specific function will be to complete the development and implementation of the CM Program Plan. The high-level elements of the CM Program Plan were developed with input from numerous CM standards and guidelines, issued by organizations such as the Department of Energy, the Department of Defense, the Institute of Nuclear Power Operations (INPO), the Nuclear Information and Records Management Association (NIRMA), the International Organization for Standardization (ISO), and the Institute for Configuration Management (ICM). The objective of the NPBU CM Program is to provide an integrated process for ensuring that the PBNP physical plant, the design and licensing basis requirements, and their documentation are synchronized. This will help ensure that activities performed by operations, maintenance, training, and engineering are conducted in accordance with the design basis requirements.

Wisconsin Electric is in the process of addressing commitments made to the NRC to correct deficiencies noted by the NRC as discussed at an enforcement conference in September 1996. We have also made a number of startup commitments, referenced in an NRC Confirmatory Action Letter, dated January 3, 1997, which will be undertaken prior to restart of Point Beach Unit 2. These commitments include reviews of testing and surveillance procedure acceptance criteria to further assure that design basis requirements are met; reviews to further assure that post-maintenance testing is properly performed; implementing improvements to the 10 CFR 50.59 and Condition Reporting processes; and reviews of open items to further assure potentially degraded equipment conditions are addressed. As an example, we have performed reviews to further assure that Inservice Testing (IST) acceptance criteria for pumps and valves meet the design basis / accident analysis requirements. Pumps and valves reviewed were found to be able to perform their design function and no operability concerns were identified. Completion of the remaining commitments will provide additional assurance that PBNP is operated, maintained, and tested consistent with the design bases.

We would be pleased to answer any questions you may have.

Sincerely,

Richard R. Grigg

Richard R. Grigg
President and Chief Nuclear Officer
Wisconsin Electric

cc: NRC Director, Nuclear Reactor Regulation
NRC Project Manager
NRC Regional Administrator
NRC Resident Inspector
Wisconsin Public Service Commission

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

Cindy L. Grenier
Notary Public, State of Wisconsin
My commission expires 8-22-99

*See attachment list on next page

Attachments:

A. Description of PBNP engineering design and configuration control processes (item (a))
B. Rationale for concluding that design bases requirements are translated into PBNP procedures (item (b))
C. Rationale for concluding that PBNP SSC configuration and performance are consistent with design bases (item (c))
D. Processes for identification of problems and implementation of corrective actions (item (d))
E. Overall effectiveness of current processes and programs in concluding that the configuration of PBNP is consistent with the design bases (item (e))
F. Design Basis Document Status
G. List of Acronyms Used

ATTACHMENT A

(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

The engineering design and configuration control processes for PBNP require that changes to PBNP are evaluated and implemented in accordance with applicable NRC regulations, including 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR Part 50. These control processes provide reasonable assurance that the PBNP design bases are appropriately considered prior to making changes to structures, systems, and components important to safety.

DESIGN AND CONFIGURATION CONTROL

NPBU procedure NP 7.2.6, "Engineering Change Process," provides overall guidance and direction for the completion of engineering changes associated with PBNP. This procedure provides a road map and direction to allow the designer to determine what design and installation procedures are required to complete an engineering change and the level of control that is necessary. The engineering change process for PBNP and its associated procedures implement ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants," as it applies to plant changes.

The primary function of the Engineering Change procedure is to provide direction for the user to categorize an engineering change into one of five types and to provide guidance as to which additional procedures and controls apply to the processing of each type of change. The five types of engineering changes are:

1. Non-QA Change: Engineering changes that are Non-Safety-Related and Non-Quality Assurance-Related are considered Non-QA changes. These changes are controlled by the requirements of the Non-QA scope sections of procedures NP 7.2.1, "Modification Request," and NP 7.2.2, "Design Control".

The following processes (types 2 through 5) involve Quality Assurance-related (QA Scope) changes and as such require preparation, review and approval by individuals formally qualified in accordance with the PBNP Engineering Support Personnel Training Program. This includes formal "Design Control" and "Modification Process" training and demonstration of competency in the task.

2. Design Change: Physical modifications to a plant system, structure or component are considered design changes. These changes are controlled in accordance with the requirements of the QA scope sections of the "Modification Request," and the "Design Control" procedures.

3. Document Change: Changes to controlled plant documents (e.g. calculations) that do not result in a physical change to a plant system, structure or component are considered document changes. These changes are controlled in accordance with procedures that exist specifically for the control of these documents, such as procedure NP 7.2.4, "Calculation Preparation, Review, and Approval".

4. Equivalent Change: Replacements of plant hardware with equivalent but not identical components are considered equivalent changes. These changes are controlled in accordance with the requirements of procedure NP 9.3.3, "Spare Parts Equivalency Evaluation".

5. Special Condition Change: Engineering changes involving special processes, such as changes to plant setpoints, changes to computer software, and temporary plant modifications are considered special condition changes. These changes are controlled by their respective procedures, such as NP 7.3.8, "Instructions for Making Changes to PBNP Setpoint and EOP Setpoint Documents," NP 1.5.2, "Computer Software and Data Management," and NP 7.3.1, "Temporary Modifications".

With the exception of engineering changes that are changes to plant documents only (Type 3), each of these processes require that the change be screened for evaluation and, if necessary, evaluated in accordance with 10 CFR 50.59 as implemented by procedure NP 10.3.1, "Authorization of Changes, Tests and Experiments" (described later in this Attachment A).

In addition, procedures exist in each of the following areas to support the engineering change process and configuration control at PBNP:

- Design Control
- Drawing Control
- 10 CFR 50.59 Reviews
- FSAR Revisions
- Calculation Creation and Control
- Field Changes to Design Output
- Computer Software and Data Management
- Control of Physical Changes to Plant Hardware
- Creation and Control of Specifications
- Setpoint Changes
- Spare Parts Equivalency Evaluations
- Temporary Modifications

As indicated above, physical changes to the design of PBNP are completed in accordance with the "Modification Requests" and "Design Control" procedures. All physical changes are defined as either a Modification Request or a Non-Mod Design Change by the Design Group Head. This determination is made based upon several factors including nuclear

safety significance, the need for a full 10 CFR 50.59 evaluation, and the QA classification of the systems, structures or components affected by the change. The Design Group Head also specifies the design controls applicable to a change.

Design change procedures include requirements that are intended to result in plant configuration control and continued conformance with the plant licensing/design bases. The following is a summary of these requirements:

1. Each safety-related design change requires completion of a design input checklist. This checklist assists the designer in ensuring that all appropriate design change inputs are considered. In particular, the designer is required to include applicable codes, standards and regulatory requirements as inputs in the design. The designer is also required to consider the basic function of each system, structure or component affected by the design change. For those design changes that have been classified as a Modification Request, an additional formal design verification is required per the requirements of the "Design Control" procedure. Both the design input checklist and design verification are reviewed by an independent technical reviewer.

2. Design changes also require the creation of a checklist called the Documentation Update Sheet. This checklist is normally created during the design process and is used to identify plant configuration, operation, and maintenance documents and procedures that may require revision due to the design change. The Documentation Update Sheet specifically requires that the FSAR, Technical Specifications, and Design Basis Documents be reviewed to determine if they are affected.

3. While not a requirement for all plant design changes, a team of individuals is normally assigned to work with a project manager for significant scope design changes. This team includes members from different plant organizations (e.g. operations, maintenance, and training) which yields a diversity and depth of input to the design process.

4. Design changes that are classified as Modification Requests require formal approval and release by the Design Group Head.

5. Design changes require the application of the "Authorization of Changes, Tests and Experiments (10 CFR 50.59 and 72.48 Reviews)" process. This process and how it relates to the engineering change process is described later in this Attachment A.

6. The implementation of all design changes is accomplished in accordance with written and approved procedures and work plans. Final approval to begin the installation process, final acceptance upon completion, and post-modification

testing is performed by the Operations group, as directed by the Duty Shift Superintendent (DSS), a licensed SRO.

Permanent changes to the plant configuration resulting from a design change are documented by the close-out process. This close-out process ensures completion of actions identified on the Documentation Update Sheet. The design control package remains open until the close-out process is complete and signed off. It should be noted that certain selected documentation updates[1] must be completed prior to testing of the configuration change by the Operations group as described in requirement 6, above. Control of plant configuration during the process of installing a design change is accomplished by the use of written and approved installation documents as described above and the use of procedure NP 1.9.15, "Danger Tag Procedure." This procedure has the dual purpose of controlling plant configuration and assuring proper isolation of plant equipment for protection of the individuals and affected equipment associated with the implementation of the change. Interim plant configurations that may exist during the installation of plant design changes are evaluated as part of the 10 CFR 50.59 process.

The "Spare Parts Equivalency Evaluation" procedure controls the replacement of plant equipment with equipment that is equivalent, but not identical, to the original. This procedure includes specific cautions to avoid inappropriate use of this process to implement design changes. In particular, it states that the "Modification Request" procedure shall be used if the function of the system, structure, or component will be changed or if changes are needed to correct design deficiencies. This process requires that the spare parts equivalency evaluation be reviewed by a second qualified engineer, that 10 CFR 50.59 reviews be applied to the change, and that a modified version of the Document Update Sheet be used to identify and track required revisions to plant documents.

"Special Condition" changes to PBNP do not result in physical changes to plant hardware, but could result in changes in the operation and function of systems, structures, or components. Examples include changes to adjustable plant setpoints and changes to computer software. These changes are controlled in accordance with procedures that have been created for the specific type of change. These procedures include the following requirements: (1) independent review of changes, (2) the application of the "Authorization of Changes, Tests and Experiments (10 CFR 50.59 and 72.48 Reviews)" process, and (3) the updating of plant documentation affected by the change.

The effectiveness of the processes described above for the implementation of engineering changes is routinely assessed by the completion of audits conducted in accordance with 10 CFR 50, Appendix B and the PBNP QA program. The PBNP QA program is

[1] Prior to post-mod testing, the project manager shall ensure that necessary Control Room, Work Control Center, and I&C documents are updated with "pen-and-ink" changes to reflect the modification. These documents include at a minimum the Control Room and Work Control Center set of P&IDs, Elementaries, Logics, and Master Data Book; and the I&C controlled set of reactor protection and engineered safeguards elementaries.

described later in this Attachment A and results of these audits are described in Attachments C.

10 CFR 50.59 PROCESS

PBNP has a formal program and process to meet the requirements of 10 CFR 50.59, "Changes, tests and experiments." Proposed changes to the PBNP facility or procedures, tests, or experiments must be evaluated to determine if an unreviewed safety question (USQ) or a change in or conflict with the PBNP Technical Specifications (Tech Specs) is involved in the proposed activity. This process requires an amendment to the PBNP operating license(s) prior to implementation of any proposed activities that involve a USQ or a change in or a conflict with the Tech Specs. The 50.59 safety evaluation program consists of a number of elements, including the following:

1. Safety Evaluation: In accordance with 10 CFR 50.59, safety evaluations are performed to determine whether proposed changes to the facility and procedures, tests, or experiments involve a USQ or a change in or conflict with Tech Specs. These evaluations are performed in accordance with procedure NP 10.3.1, "Authorization of Changes, Tests, and Experiments (10 CFR 50.59 and 72.48 Reviews)," which specifies the process, requirements, and guidance for evaluating proposed changes, tests, and experiments under 10 CFR 50.59. The procedure involves an initial screening to determine if the proposed modification, procedure change, test, or experiment is within the scope of 10 CFR 50.59 and specifies criteria to determine if a full safety evaluation is required.

2. Qualification: Personnel that perform safety evaluation screenings or prepare, review, or approve safety evaluations have received training on the current process. However, we recently determined that the training and qualification process is inconsistent among the various training programs. For example, in the engineering support training program, formal qualification with a sign-off by a qualified individual is a requirement for preparers of safety evaluation screenings and evaluations. This qualification includes verification of completion of formal 10 CFR 50.59 safety evaluation training. Operations personnel qualify to perform screenings via a qualification card using the "total card" concept. This process is being evaluated by the 10 CFR 50.59 Process Improvement team to obtain consistency relative to the training and qualification requirements for those who perform screenings and those who prepare, review, or approve safety evaluations.

3. Documentation: A Point Beach form (PBF-1515, "Nuclear Power Department Safety Evaluation Report") is used to document the 50.59 screening, safety evaluation, reviews, approvals, and the final determination as to whether a USQ or a change in or conflict with the Tech Specs is involved in the proposed change, test, or experiment. The form implements the requirements of PBNP Technical Specification 15.6.10.Q ("Plant Operating Records") and 10 CFR 50.59 to maintain a record of changes in the facility and procedures, tests, or experiments

including a written safety evaluation, which provides the basis for the determination that the change, test, or experiment does not involve a USQ. The preparer, reviewer, independent reviewers from a multidisciplinary review team, Manager's Supervisory Staff (MSS) representative, and Plant Manager sign the safety evaluation form to indicate their approval of the safety evaluation and the conclusions regarding whether a USQ or a change in or conflict with Tech Specs is involved in the proposed activity. A record of the MSS review is maintained in the MSS meeting minutes. These minutes and the completed safety evaluations on form PBF-1515 are retained in permanent plant file records.

4. Review and Approval: Following the preparation of a 50.59 safety evaluation by a qualified preparer, a qualified reviewer performs a technical review of the safety evaluation. At least two qualified individuals from a recently-added multidisciplinary review team then perform an independent review of the safety evaluation. The final review is performed by the PBNP Manager's Supervisory Staff (MSS) who reviews the safety evaluation in accordance with PBNP Tech Spec 15.6.5.1.9.c (Manager's Supervisory Staff responsibilities). The PBNP Plant Manager, or his representative, a qualified Duty and Call Superintendent, approves all safety evaluations. If the proposed change, test, or experiment involves a USQ or a change in or conflict with the Tech Specs, then the proposed activity is not authorized to be implemented until an appropriate license amendment, requested in accordance with 10 CFR 50.90, is approved by the NRC. In addition to the above reviews and approvals received prior to implementation of the proposed activity, the PBNP Off-Site Review Committee (OSRC) also reviews safety evaluations after implementation of the proposed change, test, or experiment as an independent check of the 50.59 safety evaluation process. When changes to or conflicts with Tech Specs or USQs are involved in a proposed activity, the OSRC reviews the safety evaluation and associated license amendment requests prior to submittal to the NRC.

5. Reporting: As specified in 10 CFR 50.59 and Tech Spec 15.6.9.1.B.2.C, a report containing a brief description of the changes to the facility and procedures, tests, and experiments implemented under 10 CFR 50.59 is submitted annually to the NRC. This report also provides a summary of the safety evaluations for each activity, including the basis for concluding whether or not a USQ or a change in or conflict with Tech Specs was involved in the proposed activity.

The overall purpose of the 50.59 process is to ensure that no change to the facility or procedures, tests, or experiments is implemented without prior NRC review and approval, if it involves a USQ or a change in or conflict with the Tech Specs. However, a recent violation identified by the NRC involved the failure to identify the need for a required Tech Spec change regarding a non-conforming condition on the service water system and to submit a license amendment request in a timely manner for NRC approval. Violations were also identified regarding the implementation of the 10 CFR 72.48 safety evaluation process, which is nearly identical to the 50.59 process, but is applicable to the PBNP

Independent Spent Fuel Storage Installation. These violations indicated that improvements were required in the 50.59 process utilized at PBNP.

Wisconsin Electric has committed to make short-term and long-term improvements to the 50.59 process to strengthen its effectiveness. The short-term improvements included (1) the addition of an independent review of all 50.59 safety evaluations by two members from a multidisciplinary review team prior to review by the MSS and (2) the preparation, issuance, and training on enhanced guidance and criteria for the preparation and review of 50.59 safety evaluations. Both of these short-term improvements have been implemented, procedures have been revised, and associated training completed. The long-term improvements require conducting a formal process improvement effort for the 50.59 process and completing implementation of and training on the recommended changes by May 1997.

We recently reviewed 50.59 screenings conducted in 1996 to provide added assurance that the associated modifications, procedure changes, tests, and experiments did not involve any USQs or changes in or conflict with the Tech Specs. This review, which included over four hundred 50.59 screenings, identified twenty one screenings which require a full 50.59 evaluation and another four which require additional detail. The initial assessments of these items has indicated that none of them involve an unreviewed safety question or a change in or conflict with the Tech Specs.

10 CFR 50.71(e) PROCESS

The NPBU policy for the maintenance and update of the PBNP FSAR in accordance with the requirements of 10 CFR 50.71(e) is addressed by Administrative Manual AM 3-14 "FSAR Maintenance and Update Policy". This policy explains the regulatory requirements for the content and updating of the FSAR; NPBU management expectations for the FSAR; the responsibilities of NPBU personnel to update and maintain the information in the FSAR; and activities performed to ensure the FSAR meets NPBU expectations and NRC regulations. It emphasizes that, since the PBNP FSAR is used by NPBU personnel and the NRC in safety significant activities, the accuracy, timeliness, and completeness of the information in the FSAR is essential to ensure the conclusions of these activities are correct and appropriate. AM 3-14 also states that it is the responsibility of all NPBU personnel who use the FSAR, to assist with the update and maintenance of the FSAR, or whose activities affect the FSAR, to maintain the FSAR in accordance with NRC regulations.

Procedure NP 5.2.6 "FSAR Revisions" describes the process in NPBU to periodically revise the PBNP FSAR in accordance with 10 CFR 50.71(e). The responsibility for compiling changes to the FSAR and preparing the annual FSAR update is assigned to the FSAR Coordinator in the NPBU Licensing group. Annual FSAR updates are issued in June of each year, reflecting changes to the facility which were effective at the end of the previous year, consistent with the requirements of 10 CFR 50.71(e).

Any individual in the NPBU who conducts an activity that affects the content of the FSAR is responsible for evaluating the effects on the FSAR and initiating an FSAR Change Request (FCR). The evaluation of the effects on the FSAR includes a thorough check of the FSAR (electronically or with a hard copy) to identify all FSAR sections affected by the proposed change, and a check to ensure that a 10 CFR 50.59 safety evaluation was completed, if necessary, for the activity resulting in the FSAR change. The completed FCR contains reference documents for the FSAR change, a marked-up copy of affected sections of the FSAR, and a description of the evaluation supporting this change. The FCR is reviewed by an independent individual competent in the subject of the FCR, and then forwarded to the FSAR Coordinator. The FSAR Coordinator reviews the FCR to ensure it is appropriate and complete, and if so, accepts it and assigns a number for tracking and dispositioning. The FSAR Change Request is generally prepared and submitted when the FSAR change is identified. However, to support the annual June update, it is required that FSAR Change Requests be submitted to the FSAR Coordinator by February 15th for any FSAR changes due to activities completed in the previous calendar year.

To prepare the annual update, the FSAR Coordinator compiles the FSAR changes submitted to form a change package made up of FSAR replacement pages. The change package, accompanied by a description of the changes, is submitted to all NPBU Sections and specifically to the system engineers for review. NP 5.2.6 contains guidance and expectations for this change package review. The FSAR Coordinator reviews and incorporates the comments from the change package review into the final update, and coordinates distribution of the final update internally and to the NRC.

Recent inspections by the NRC have identified examples of past weaknesses in implementing the 10 CFR 50.71(e) requirement for the PBNP FSAR. We recently received a violation for three examples of failure to update the FSAR. In addition, we recently identified an increasing number of Condition Reports (the Condition Report process is described in Attachment D) pertaining to FSAR inaccuracies, which indicated a negative trend in the FSAR maintenance process. In response to these findings, we chartered an interdisciplinary team to review FSAR Chapter 9, "Auxiliary and Emergency Systems", and a separate process improvement team to assess our existing FSAR update process.

The FSAR Chapter 9 review team and the process improvement team found that our FSAR maintenance and update process was not well-integrated into routine work processes, and that personnel awareness of the need to keep the FSAR accurate and up-to-date required improvement. FSAR maintenance was encompassed in one procedure, NP 5.2.6, "FSAR Revisions," and related procedures, such as the Technical Specification Change Request procedure, did not prompt the FSAR to be reviewed and appropriate FSAR updates to be submitted.

The FSAR Chapter 9 review team identified some generic weaknesses in the configuration control process with respect to FSAR updating, including: (1) equipment was being abandoned without changing the FSAR description, (2) processes were being abandoned or revised without changing the FSAR description, and (3) the original FSAR contained errors that have not been corrected. In most of these cases, the actual plant configuration or process had been adequately evaluated as required by 10 CFR 50.59, but the appropriate FSAR updates had not always been made. Also, the FSAR update process improvement team identified weaknesses in NP 5.2.6 in that it did not provide a means for formally documenting proposed FSAR changes with their basis, and that there was no formal means for tracking and appropriately dispositioning the FSAR changes which were submitted.

As a result, AM 3-14, "FSAR Maintenance and Update Policy" was prepared, and NP 5.2.6 was revised, both in October 1996. The current FSAR update process described above reflects these recent changes to NP 5.2.6. Computer-based word-search tools for querying the FSAR and Tech Specs have been provided to facilitate identification of all FSAR effects associated with a change.

The overall philosophy for maintaining the PBNP FSAR to meet the requirements of 10 CFR 50.71(e) can be described by a defense-in-depth approach, as follows:

1. The first defense is awareness by NPBU personnel of the importance of the FSAR and the various means by which it could be affected. This is being accomplished by communication by NPBU management, the issuance of AM 3-14 and the revision to NP 5.2.6. In addition, a computer-based training module is currently being developed specifically for this purpose. This module will provide FSAR awareness and overview training to all NPBU personnel. More specific and practical training for selected NPBU groups is also being planned for 1997.

2. The second defense is provided by procedures which prompt an evaluation of any effects on the FSAR when changes to the facility occur. The design control process, Technical Specification change process, and many other processes can routinely affect the FSAR. These other processes include: (1) procedure revisions, (2) calculations, (3) setpoint changes, (4) safety evaluations, and (5) organization changes. Procedures governing these processes are being reviewed and revised as appropriate to provide the appropriate prompts. NP 5.2.6 revisions have been completed.

3. The third defense is provided by the Manager's Supervisory Staff (MSS). The MSS provides the awareness, knowledge, and experience to assure that the effects of all changes to the plant are appropriately evaluated.

4. In the event that an FSAR change is not appropriately identified during the conduct of an activity which affects the FSAR, the final defense is provided by the

Licensing group. On an annual basis, the Licensing group performs a review of the safety evaluations, NRC correspondence, and administrative or organizational changes of the previous year to identify any effects on the FSAR that may have been overlooked.

We believe that this defense-in-depth approach provides reasonable assurance that the required FSAR changes are identified and submitted within the timeframe required by 10 CFR 50.71(e). To provide further assurance that the PBNP FSAR complies with the requirements of 10 CFR 50.71(e), we are developing plans and a schedule for a complete review and update of the FSAR as previously stated in WE to NRC letter VPNPD-96-096. The review of FSAR sections will be prioritized based on Probabilistic Safety Assessment (PSA) insights and will be coordinated with ongoing initiatives at PBNP. Periodic FSAR updates (as described above) will be issued during this review. We expect to complete this review and update by June 1998. This schedule is consistent with our present update schedule and with the revision to the NRC enforcement policy dated October 18, 1996.

<u>APPENDIX B TO 10 CFR 50 WITH RESPECT TO DESIGN AND CONFIGURATION CONTROL PROCESS</u>

Wisconsin Electric has established and implemented a Quality Assurance Program in accordance with the criteria specified in 10 CFR 50, Appendix B, as required by 10 CFR 50.54(a)(1). This program is described in Section 1.8 of the PBNP FSAR, as required by 10 CFR 50.34(b)(6)(ii). Changes to this program description are required to be submitted to the NRC pursuant to 10 CFR 50.54(a)(3).

Wisconsin Electric is committed to the guidance provided in ANSI N18.7-1976 for its Quality Assurance Program except as specifically noted in Section 1.8 of the PBNP FSAR. Either complete or partial commitment, with alternative methods discussed in the FSAR, is made to the following ANSI standards, which relate to the design and configuration control processes:

- ANSI N45.2.4-1972, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
- ANSI N45.2.5-1974, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"
- ANSI N45.2.8-1975, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

- ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

The Quality Assurance Program is further delineated by written policies, procedures, or instructions, which include the NPBU Administrative Manual, the NPBU Procedures Manual, and the NPBU Organization Manual. The requirements of the Quality Assurance Program and the various ANSI standards to which it commits are incorporated into these detailed implementing procedures. The design and configuration control processes are described in these documents, including the organization, responsibilities and the quality assurance requirements for these processes.

The Quality Assurance Section (QAS) is responsible for assuring that an appropriate quality assurance program is established and effectively executed, and verifying that activities affecting safety-related functions have been correctly performed. This organization has sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. The QAS reports directly to the Chief Nuclear Officer and is provided the required authority, organizational freedom, including independence from cost and schedule when opposed to safety considerations. Members of QAS have direct access to appropriate levels of management as may be necessary to perform their functions.

Sufficient audits are to be performed by the QAS to meet the requirements of Section 4.5 of ANSI N18.7-1976 as implemented by procedure NP 11.2.3, "Internal Assessment Program Coverage, Planning, Scheduling, and Reporting". While the scheduling of these audits is performance-based, an audit of all safety-related functions is completed within a maximum period of two years, unless otherwise specified in the PBNP Technical Specifications or regulations. As such, all of the criteria of 10 CFR 50, Appendix B, are required to be audited at least once every two years. Each audit report lists the criteria in whole or in part covered by the assessment. A cumulative listing is maintained to assure that all of the criteria of 10 CFR 50, Appendix B are audited at least once every two years, including those associated with design and configuration control.

Audits are performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the area being audited. Audits are conducted in accordance with the provisions of ANSI N45.2.12-1977 "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants," and led by individuals certified as Lead Auditors in accordance with the requirements of ANSI N45.2.23-1978 "Qualifications of Quality Assurance Program Audit Personnel for Nuclear Power Plants" as implemented in procedures NP 11.2.1, "Internal Assessments," and NP 11.1.1, "Lead Auditor Qualifications," respectively. Audit results are documented and reviewed by management personnel having responsibility in the area audited. Follow-up action, including re-audit of deficient areas, is taken when indicated. Further information regarding the audit process is provided in Attachment D of this letter.

Periodic audits of the design and configuration control processes are conducted by the QAS to assess compliance of all aspects of the quality assurance program and to determine the effectiveness of the program. Where practical, these audits utilize performance based techniques, such as direct observation of field activities, verification of as-built conditions, design bases, configuration control, and the technical adequacy of documents and procedures. As an example, periodic operational readiness assessments of safety-related plant systems are conducted using vertical-slice techniques (based on NRC Safety System Functional Inspection (SSFI) guidance) and utilizing technical specialists with expertise on the subject matter. Further information regarding the results of these vertical slice audits is provided in Attachments B and C of this letter.

ATTACHMENT B

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

There are several programs in place at Point Beach which assess and/or maintain the consistency between the operation, maintenance, and testing of PBNP with the design bases. These activities include:

- WE Assessments;
- Design Basis Document Validation;
- Instrument Setpoint Verification;
- 10 CFR 50.59 Reviews; and
- External Assessments.

In addition to these activities, the following recent initiatives are currently being implemented to further evaluate and enhance the operation, maintenance, and testing of the plant with respect to the design bases:

- Inservice Testing Program Enhancements;
- Post-Maintenance Testing Process and Procedures;
- Maintenance and Instrument & Control Procedure Upgrade Project; and
- Operations Procedure Upgrade Project.

Each of these programs and processes is described below or elsewhere in this letter. The results of WE assessments (e.g. Vertical Slice Audits), DBD validations, the instrument setpoint verification program, procedure enhancement programs, and external assessments (such as NRC inspections) provide an indication of the effectiveness of translating design basis requirements into operating, maintenance, and testing procedures. Based upon this objective evidence, reasonable assurance exists that design basis requirements have been translated into operating, maintenance, and testing procedures.

Recently identified instances where procedures were found to be deficient may indicate weaknesses in configuration management of procedures relative to design bases. PBNP has relied primarily on the 10 CFR 50.59 safety evaluation process and the experience and qualifications of the persons preparing, reviewing, and approving changes to procedures to ensure that design bases are maintained in our procedures. As discussed in this section, several recent initiatives have been completed and others are ongoing in an effort to strengthen the link between design basis information and operation, maintenance, and testing procedures. In addition, a group has been added to the organizational structure of NPBU whose specific function will be to complete the development and implementation of the NPBU Configuration Management (CM) Program Plan. The objective of the NPBU CM Program is to provide an integrated process for ensuring that the PBNP physical plant, the design and licensing basis requirements, and their documentation are

synchronized. This will help ensure that activities performed by operations, maintenance, training, and engineering are conducted in accordance with the design basis requirements.

WE ASSESSMENTS

1. Vertical Slice Audits - Since 1988, PBNP has conducted voluntary internal assessments of eight selected systems and functional areas using techniques similar to those used in Safety System Functional Inspections conducted by the NRC. These assessments are referred to as Vertical Slice Audits (VSAs). VSAs focus on the functionality of components in the selected PBNP safety system by verifying that the current configuration and performance are consistent with the design and licensing bases. The following areas are evaluated during this audit: (1) system and equipment design, including modifications to existing equipment design, specification of purchased components, QA-scoping of SSC, original design bases, and final design documentation, (2) test and surveillance of the system performance to verify that the SSC meet their intended function, (3) maintenance of the system with a focus on preserving the ability to perform the safety functions, (4) operation of the system with an emphasis on operating the system within the design bases, (5) corrective action and operating problems, and (6) additional areas of training, document and record control, plant status control, and other supporting areas.

   Since their inception in 1988, the following VSAs have been completed:

   - Emergency Diesel Generator System (EDG), (audit report number A-SP-88-02, performed 1/4-3/14/88)
   - Residual Heat Removal System (RHR), (audit report number A-P-88-10, performed 8/8-10/19/88)
   - Containment Structure and Containment Spray System (CS), (audit report number A-P-89-12, performed 9/5-10/20/89)
   - Auxiliary Feed Water System (AFW), (audit report number A-P-90-12, performed 9/5-10/22/90)
   - Reactor Protection System (RPS), (audit report number A-P-91-10, performed 7/6-8/14/92)
   - Service Water System (SW), (audit report number A-P-93-01, performed 2/1/-3/12/93)
   - Instrument & Control Audit, (audit report number A-P-94-01, performed 1/17-2/25/94). This assessment was performed in lieu of an NRC SBICI per the guidelines/criteria defined in NRC procedure 93807. A follow-up NRC inspection was conducted.
   - Chemical & Volume Control System (CVCS), (audit report number A-P-96-02, performed 1/22-3/1/96)

Each of these internal vertical slice audits concluded that the system being audited was capable of performing its safety-related functions[1]. These audits have, for the most part, concluded that operating procedures were satisfactory. Many of the identified weaknesses in operating procedures were related to the level of detail and guidance contained in these procedures (this is particularly true with the earlier VSAs). As discussed later in this Attachment B, the Operations Procedure Upgrade Project (initiated in 1993) was initiated to improve the human factors usage and technical content of PBNP Emergency Operating Procedures (EOPs) and Abnormal Operating Procedures (AOPs). With respect to maintenance and testing procedures, these audits generally concluded that testing and maintenance of the system are adequate to demonstrate that the system would perform its intended functions. However, these audits have identified weaknesses related to testing acceptance criteria and Post Maintenance Testing. As noted in this Attachment B under "Inservice Testing Program Enhancements" and "Post-Maintenance Testing Process and Procedures", specific corrective actions have been completed and others are underway to review and, if necessary, revise the test acceptance criteria and post-maintenance testing procedures.

The validity and effectiveness of our VSAs has been confirmed by two NRC inspections. An NRC Service Water System Operational Performance Inspection (SWSOPI) considered our Service Water System Vertical Slice Audit (A-P-93-01) to have been a significant effort since a number of the weaknesses and deficiencies identified during their inspection had been previously addressed by the audit. An NRC follow-up inspection of the Instrument & Control Audit (A-P-94-01) concluded that the internal audit was comprehensive, that the proposed short- and long-term corrective actions were reasonable, and that our operability determinations were adequate.

2. Self-Assessment on Design Basis / Plant Configuration Consistency - A recently completed limited self-assessment (report number S-A-97-01) evaluated the effectiveness of our processes for controlling the PBNP configuration consistent with the design bases. The self-assessment was based on design parameters contained in DBDs for the Reactor Coolant System and the 125 VDC System. The Reactor Coolant System (RCS) was chosen as one of the systems because a DBD had recently been generated for this system and neither a vertical slice nor a design audit had previously been performed on the system. The 125VDC system was the other system selected because the original DBD had recently been updated. This provided the self-assessment team an opportunity to assess process and technical enhancements originating from past audits and design related nonconformances against well documented design basis information. The scope of system attributes (components

[1] During the evaluation of the service water system (audit SWS, A-P-93-01), the audit team could not reach a conclusion of the ability of the system to perform its intended safety function under all possible design basis scenarios. Subsequent resolution of the associated condition report on this issue found that the scenario in question did not reflect a design bases condition.

and related operation) reviewed was intentionally selective so a vertical slice assessment could be performed.

With respect to operating, maintenance, and testing procedures, the overall results of the self-assessment indicated that the translation of design, design changes and vendor specifications into procedures was somewhat inconsistent. It was also concluded that procedures provided adequate instruction to experienced users, but were weak in providing detailed step-by-step instructions, accurate component identification and equipment status. While a number of activities such as procedure upgrades and Inservice Testing evaluations had been initiated, the self-assessment team has recommended a further in-depth review of procedures. This includes review of processes relied on for the translation of design information into procedures and reviews of industry best practices for procedure format and content. Attachment C discusses the results of this self-assessment as it relates to plant configuration and performance.

DESIGN BASIS DOCUMENT VALIDATION

The purpose of DBD validation is to provide reasonable assurance that design basis attributes are properly and consistently implemented in the physical plant and in those documents important for the support of plant operation. DBD validation includes a limited review of operating, maintenance, and testing procedures against the design bases. As an example, the validation of a system or component performance requirement may involve reviewing test acceptance criteria to determine if the allowable value(s) for the performance parameter are within the boundaries of the design bases. Refer to Attachment C for further discussion on the DBD validation process.

As identified in Attachment F, DBD validations have been completed on 21 systems/topical areas and will continue as new DBDs are written. To date, of the over 500 design basis "attributes" validated, approximately 20% have included a review of plant operating, maintenance, and testing procedures related to the particular attribute. Validation of test acceptance criteria and test results has shown that actual test data has demonstrated conformance with the design bases. However, some attributes were identified where the acceptance criteria limits were not conservative with respect to design basis requirements. Specific corrective actions are now underway to review and, if necessary, revise the test acceptance criteria to be consistent with the design bases. Refer to the section in this Attachment B under "Inservice Testing Program Enhancements" for additional discussion on these actions. DBD validation related to operating procedure reviews have shown that although procedural improvements could be made, the procedures are generally consistent with the design bases.

INSTRUMENT SETPOINT VERIFICATION

PBNP initiated an Instrument Setpoint Verification Program in 1993 to formally calculate the minimum required margins between safety-related instrument setpoints, their analytical limits, and Tech Spec values, taking into account the appropriate instrumentation uncertainties and process measurement effects. ISA Standard 67.04-1994 was used as guidance for the setpoint methodology developed for these setpoint calculations. Many of the analytical limits used in these calculations were researched and documented in conjunction with the PBNP DBD program.

To date, setpoint margins associated with fifteen different reactor trip, safeguards actuation, and other Tech Spec variables have been calculated and all were found to contain sufficient margins from their associated analytical limits to the existing field and Tech Spec settings. Therefore, no setpoint changes have been required. The PBNP Instrument Setpoint Verification program was independently reviewed during a 1994 WE Instrument & Control Audit performed in lieu of an NRC Systems Based Instrument and Control Inspection (SBICI). This audit found both strengths and weaknesses associated with the program. A follow-up NRC inspection noted that the audit observations relating to the Instrument Setpoint Verification Program were being adequately addressed.

10 CFR 50.59 REVIEW OF PROCEDURE CHANGES

Operating, maintenance, and testing procedures at PBNP are currently screened for 10 CFR 50.59 applicability at the time of creation and at subsequent temporary or permanent revisions. NPBU uses the 50.59 evaluation process (Attachment A describes our 50.59 evaluation process in more detail) to direct the procedure change author to consult the design basis information contained in the FSAR and current licensing basis regarding the effect of the change.

A comprehensive reorganization of PBNP administrative procedures occurred in 1994-1995. As part of this effort, the link between the associated procedural control procedures and the 50.59 safety evaluation process was enhanced to ensure that procedures are pre-screened for 50.59 applicability when created or revised.[2] The 50.59 "pre-screening" examination of a procedural change can result in a requirement to perform a 50.59 screening, a full 50.59 safety evaluation, or can conclude that a 50.59 screening is not warranted. This tiered approach is designed to efficiently consider procedural changes in light of the 50.59 regulation to ensure that procedures are reviewed commensurate with their importance to safety. As an example, a change to a procedure which is purely administrative in nature would be pre-screened out as not requiring further examination under 50.59. However, a technical change to an

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[2] Work or operations conducted in accordance with guidance documents do not receive the scrutiny of a 50.59 review. Typically, work done in accordance with guidance documents is simple in nature but may not be limited to non-safety related SSCs. The 50.59 process improvement team (discussed below) is evaluating this issue and is considering when 50.59 reviews should be applied to the work processes governed by guidance documents.

operating procedure would require completion of the 50.59 screening form, where a decision regarding the need for a full evaluation under 50.59 would be made and documented.

Manager's Supervisory Staff (MSS) review of 50.59 safety evaluations for possible USQs is directed by the PBNP Tech Specs. Additionally, the PBNP Tech Specs discern between Major and Minor technical procedures, and which of these procedures require MSS review. The procedure change process, however, is broader than the narrowly defined Tech Spec requirements, and provides a structured, umbrella framework to examine changes in accordance with 10 CFR 50.59.

Findings by PBNP's Off-Site Review Committee (OSRC) and several condition reports identified problems with 10 CFR 50.59 implementation and prompted us to develop a 50.59 improvement action plan. Prior to the implementation of this action plan, routine NRC inspection reports in 1996 identified deficiencies related to the 50.59 review process, and also prompted the formation of a formal process improvement team. The goal of the team is to re-examine and re-define the 50.59 process at PBNP, taking into account the re-examination of 50.59 by the NRC and best practices from the industry. Recommended process changes are to be implemented by May 1997. Completed short-term corrective actions include an additional independent review by two members of a multi-disciplinary review team (composed of acknowledged 50.59 process experts in NPBU) and implementation of additional management guidance in the area of safety evaluation preparation and review.

We recently reviewed 50.59 screenings conducted in 1996 to provide added assurance that the associated modifications, procedure changes, tests, and experiments did not involve any USQs or changes in or conflict with the Tech Specs. This review, which included over four hundred 50.59 screenings, identified twenty one screenings which require a full 50.59 evaluation and another four which require additional detail. The initial assessments of these items has indicated that none of them involve an unreviewed safety question or a change in or conflict with the Tech Specs.

INSERVICE TESTING PROGRAM ENHANCEMENTS

A weak link between the design bases and the inservice testing acceptance criteria has been identified as a deficiency by DBD Validations, WE QA assessments, and NRC inspections. Accordingly, commitments were made to the NRC in September 1996 involving several short-term and long-term corrective actions which affect testing procedures.

The Inservice Test series of procedures implement pump and valve testing requirements of ASME Section XI. The problem identified was that while the Inservice Test procedures comply with Section XI limits for establishing acceptance criteria, the acceptance criteria did not, in all cases, represent design basis limits for the particular

component. Accordingly, the inservice testing acceptance criteria for pump and valve testing have been reviewed against their design bases.

The review included a comparison of design bases to test acceptance criteria and to actual component test data. The test data have demonstrated that both pump and valve components have performed within their design bases. While some test procedures and acceptance criteria have been modified as a result of this effort, this was done so that the test procedures and the acceptance criteria would properly reflect design basis criteria (although the test acceptance criteria may, in some cases, actually be more conservative than what the design bases require). Based on the review results, historical pump and valve performance has conformed to the design bases, and the inservice testing acceptance criteria, as modified, will continue to conservatively conform to the plant design bases.

A review of other safety-related equipment, including heat exchangers, is currently underway to ensure that inservice testing acceptance criteria reflect design basis requirements. Where acceptance criteria in existing surveillance and testing procedures are not adequately linked to the design bases, the procedure will be revised. A periodic callup and procedure is also being implemented to establish an annual review of the inservice testing program against plant design basis changes. Together with the 50.59 process, these enhancements will provide additional assurance that Inservice Test procedure acceptance criteria are being maintained conservative with respect to the design bases.

MAINTENANCE AND INSTRUMENT & CONTROL PROCEDURE UPGRADE PROJECT

The Procedure Upgrade Project (PUP) was initiated in 1991 in response to a 1990 INPO finding on Maintenance and Instrumentation & Controls (I&C) work activities. The scope of the PUP is to examine Routine Maintenance Procedures (RMPs), I&C Procedures (ICPs), Maintenance Instructions (MIs) and Maintenance Work Plans (MWPs) for technical content, and redesign them in accordance with a procedure writer's guide. This process examines design basis information from the FSAR and Tech Specs in its look at these procedures and guidance. RMPs and ICPs are considered "procedures" by the PBNP definition, and hence their change process is linked to the 50.59 process. MIs and MWPs are considered "guidance," and are not currently considered by the 50.59 process. The project is scheduled to be completed in 1998, although an extension beyond this date may be necessary to ensure that appropriate procedures are addressed. So far, the reviews conducted by this project have not found inconsistencies between these documents and design basis information contained in the FSAR and Technical Specifications.

OPERATIONS PROCEDURE UPGRADE PROJECT

The Operations Procedure Upgrade Project was initiated in 1993 to improve the human factors usage and technical content of PBNP EOPs and AOPs. The EOPs and AOPs are being upgraded in order to incorporate operator feedback and lessons learned from the simulator training. This project includes a verification of the procedures against the design basis requirements as identified in the FSAR and Technical Specifications. Selected AOPs are being converted to two-column format to be consistent with the EOPs. The EOPs have been rewritten and completion of AOP rewrites is expected in March 1997. So far, the verification conducted by this project has identified only one instance where an EOP was not consistent with information contained in the FSAR. This inconsistency was considered to be an isolated case and not safety significant.

POST-MAINTENANCE TESTING PROCESS AND PROCEDURES

Due to shortcomings identified in a WE assessment (audit report number A-P-96-15) and NRC inspection in the area of post-maintenance testing (PMT) within the last six months, NPBU formed a post-maintenance testing review group, comprised of members from Maintenance, Operations and Engineering. For work orders, "return to service" testing is proposed, examined, and reviewed by a member of the group from each of the three areas. An additional review is conducted by an Inservice Inspection (ISI) group representative for work involving equipment governed by the scope of ASME Section XI. A "Return to Service Testing Review" form accompanies the work order and documents the reviews. These reviews were begun in September, 1996, with commencement of the Unit 2 Steam Generator Replacement Outage. These reviews are intended to provide additional assurance that post-maintenance, operability, and surveillance testing requirements are properly implemented and conducted such that components/system which are returned to service are capable of performing their design functions.

To ensure that PMT performed prior to these reviews were adequate, we are currently reviewing 20% of the maintenance work orders performed since January 1, 1995 on Unit 2 or common PSA safety significant systems to verify that adequate PMT was performed to ensure system/component safety function. Additionally, we are reviewing and revising (if necessary) operating procedures that contain maintenance activities to ensure PMT and Quality Control are properly addressed by those procedures.

EXTERNAL ASSESSMENTS

Many external assessments, such as NRC inspections, have evaluated issues pertaining to design bases implementation. For example, the objective of Safety System Functional Inspections (SSFIs) is to verify the functionality of selected safety systems by inspecting the performance-related attributes of the system with a focus on the body of plant-specific

design basis information. With respect to maintenance and testing procedures, NRC inspections have identified weaknesses with testing and maintenance procedures, many of which suggest a weak link between the design bases and the inservice testing acceptance criteria. NRC inspections have also found some weaknesses related to Post Maintenance Testing. As previously identified in this Attachment B, similar weaknesses have also been identified by DBD validations and WE assessments. Specific corrective actions have been completed and others are underway to review and, if necessary, revise test acceptance criteria and Post Maintenance Testing procedures. Refer to the sections in this Attachment B under "Inservice Testing Program Enhancements" and "Post-Maintenance Testing Process and Procedures" for additional discussion on these actions.

ATTACHMENT C

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

As discussed in Attachment A, Point Beach has several engineering design and control processes which provide reasonable assurance that PBNP design bases are appropriately considered prior to making changes to systems, structures, and components important to safety. In addition, there are several programs employed at PBNP to assess and/or maintain system, structure, and component (SSC) configuration and performance against the design bases. A principal means of evaluating consistency with the design bases is the validation portion of the DBD Program. The validations are performed as an integral part of the DBD development process.

In addition to DBD validation, other programs evaluate and/or maintain SSCs for design bases consistency. These include the following:

- WE Assessments;
- As-Built Drawing Upgrade Project;
- Instrument Setpoint Verification;
- Testing and Surveillance Programs;
- Environmental and Seismic Qualification Reviews;
- QA Classification Process / Q-List; and
- External Assessments.

Each of these programs and processes is described below or elsewhere in this response. The results of DBD validation, WE assessments, as-built drawing program, instrument setpoint verification, testing and surveillance programs, environmental and seismic qualification reviews and external assessments provide an indication of the consistency of SSC configuration and performance with the design bases. Based upon this objective evidence, reasonable assurance exists that PBNP SSC configuration and performance is consistent with the design bases.

DESIGN BASIS DOCUMENT VALIDATION

The purpose of the PBNP Design Basis Document (DBD) Validation, as stated in the DBD Program Manual, is to provide reasonable assurance that the design bases attributes are properly and consistently implemented in the physical plant and in those documents important for the support of plant operation. For a given system, the DBD validation is intended to provide reasonable assurance that: (1) the system design is implemented such that the system can accomplish its functions and meet its performance requirements; (2) the system is adequately tested to demonstrate that it will accomplish its functions within its performance requirement limits; and (3) the system is properly operated during normal

and accident conditions consistent with its design bases. Similarly for topical DBDs, the validation should provide reasonable assurance that the plant is configured, tested, and operated consistent with the topical area design bases.

The DBD validation consists of both document reviews and physical walkdowns. Document reviews compare design basis "attributes" to controlled plant documents, such as drawings, calculations, equipment specifications, operating procedures, test procedures, and other procedures. Physical walkdowns compare design basis attributes to the actual plant configuration. System-level performance requirements associated with system-level functions are validated. Other selected system-level design, operational, inspection and testing, and component-level performance requirements are also validated based on the attribute's relative importance to system operation and performance. For a topical DBD, applicable types of DBD information for the topical area are validated against a representative sample of systems, structures, and components to which the topical area applies. The validation sample size is intended to be large enough for the validation to reasonably conclude that the selected design basis attributes are properly implemented in the physical plant and in supporting documents.

The validation of an attribute is considered "Acceptable" when the validation team has sufficient evidence to conclude that the design basis attribute has been properly implemented in the physical plant or in the important plant documents, or some combination of both. Otherwise, the validation attribute is "Open". Open validation attributes are reviewed for safety significance and if there is potential impact on safety, operability, or reportability, a Condition Report is generated to evaluate the item further with the appropriate level of attention.

Consistent with an overall change in the handling of DBD Open Items, other open validation attributes become DBD Open Items. In addition, the open attributes will be reviewed by the DBD group with an active SRO and the responsible System Engineer to ensure safety, operability, and reportability implications are addressed prior to being finalized as open items in the DBD, Revision 0. See the cover letter for additional discussion on how DBD Open Items are processed.

As identified in Attachment F, DBD validation has been completed on 21 systems/topics and will continue as new DBDs are written. For these DBDs, over 500 design basis attributes were validated. A review of completed DBD validation reports indicated that:

- During validation, no SSC was found to be in a condition outside the design bases requiring a 10 CFR 50.72 and/or 10 CFR 50.73 report.[1]

[1] Our DBD program has identified a few instances (outside of DBD validations) where an SSC was considered to be potentially outside its design bases requiring a 10CFR50.72 and/or 10CFR50.73 report. The limited number of these cases and diverse nature of the technical issues, compared to the large number of systems/topical design and performance attributes which were researched and validated, indicates that these are isolated instances and not representative of a generic problem with design basis conformance.

- A limited number of instances have been identified where the plant configuration, testing, or operation does not support a documented design bases, resulting in the issuance of a Condition Report and/or DBD Open Item. DBD Open Items have been recently re-reviewed for safety significance and operability / reportability by an active SRO and System Engineering personnel. Condition Reports and prompt operability determinations were prepared as deemed appropriate.
- Validation of test acceptance criteria and test results has shown that actual test data has demonstrated conformance with the design bases. However, some attributes were identified where the acceptance criteria limits were not conservative with respect to design basis requirements. Specific corrective actions are now underway to review and, if necessary, revise the test acceptance criteria accordingly to be consistent with the design bases. Refer to the section in this Attachment C under "Testing and Surveillance Programs" for additional discussion of these actions.
- Validation of a some design basis attributes could not be completed due to missing information, resulting in the validation attribute being open. These attributes have been reviewed for safety significance and, as appropriate, a Condition Report and/or DBD Open Item was issued.

WE ASSESSMENTS

Assessments are conducted by organizations within the Nuclear Power Business Unit. The most significant of these assessments as they relate to design basis issues are PBNP internal Vertical Slice Audits (VSAs). VSAs are voluntary audits of selected systems or functional areas using techniques similar to those used in Safety System Functional Inspections conducted by the NRC. In addition to Vertical Slice Audits, other assessments which have assessed plant configuration and performance include: (a) a recently completed self-assessment on design basis / plant configuration consistency; and (b) Outage Modification and Programmatic Audits which assess the engineering design and configuration control processes.

1. Vertical Slice Audits (VSAs) - VSAs focus on the functionality of components in the selected PBNP safety system by verifying that the current configuration and performance are consistent with the design and licensing bases. The following areas are evaluated during this audit: (1) system and equipment design, including modifications to existing equipment design, specification of purchased components, QA-scoping of SSC, original design bases, and final design documentation, (2) test and surveillance of the system performance to verify that the SSC meet their intended function, (3) maintenance of the system with a focus on preserving the ability to perform the safety functions, (4) operation of the system with an emphasis on operating the system within the design bases, (5) corrective action and operating problems, and (6) additional areas of training, document and record control, plant status control, and other supporting areas.

   Since their inception in 1988, the following VSAs have been completed:

- Emergency Diesel Generator System (EDG), (audit report number A-SP-88-02, performed 1/4-3/14/88)
- Residual Heat Removal System (RHR), (audit report number A-P-88-10, performed 8/8-10/19/88)
- Containment Structure and Containment Spray System (CS), (audit report number A-P-89-12, performed 9/5-10/20/89)
- Auxiliary Feed Water System (AFW), (audit report number A-P-90-12, performed 9/5-10/22/90)
- Reactor Protection System (RPS), (audit report number A-P-91-10, performed 7/6-8/14/92)
- Service Water System (SW), (audit report number A-P-93-01, performed 2/1/-3/12/93)
- Instrument & Control Audit, (audit report number A-P-94-01, performed 1/17-2/25/94). This assessment was performed in lieu of an NRC SBICI per the guidelines/criteria defined in NRC procedure 93807. A follow-up NRC inspection was conducted.
- Chemical & Volume Control System (CVCS), (audit report number A-P-96-02, performed 1/22-3/1/96)

Each of these internal vertical slice audits concluded that the system being audited was capable of performing its safety functions[2]. For the most part these audits also concluded that the design of the electrical, mechanical, and Instrument & Control portions of the system being audited was adequate and sufficient to perform their required safety functions. In general these audits also found that the testing and maintenance of the system were adequate to demonstrate that the system would perform its intended functions. However, these audits identified weaknesses with testing and maintenance procedures, primarily related to a lack of a strong link between the design bases and the inservice testing acceptance criteria. As discussed in Attachment B under "Inservice Testing (IST) Program Enhancements", a review of inservice testing results indicate that the SSC's tested performance has been consistent with the design bases.

The validity and effectiveness of our VSAs has been confirmed by two NRC inspections. An NRC Service Water System Operational Performance Inspection (SWSOPI) considered our Service Water System Vertical Slice Audit (audit report number A-P-93-01) to have been a significant effort since a number of the weaknesses and deficiencies identified during their inspection had been previously addressed by our audit. An NRC follow-up inspection of the Instrument & Control Audit (audit report number A-P-94-01) concluded that the internal audit was comprehensive, that

[2] During the evaluation of the service water system (audit SWS, A-P-93-01), the audit team could not reach a conclusion of the ability of the system to perform its intended safety function under all possible design basis scenarios. Subsequent resolution of the associated condition report on this issue found that the scenario in question did not reflect a design bases condition.

proposed short- and long-term corrective actions were reasonable, and that our operability determinations were adequate.

2. Self-Assessment on Design Basis / Plant Configuration Consistency - A recently completed limited self-assessment (report number S-A-97-01) evaluated the effectiveness of our processes for controlling the PBNP configuration consistent with the design bases. This self-assessment was based on design basis parameters contained in DBDs for the Reactor Coolant System and the 125 VDC System. The Reactor Coolant System (RCS) was chosen as one of the systems because a DBD had recently been generated for this system and neither a vertical slice nor a design audit had previously been performed on the system. The 125VDC system was the other system selected because the original DBD had recently been updated. This provided the self-assessment team an opportunity to assess process and technical enhancements originating from past audits and design related nonconformances against well documented design basis information. The scope of system attributes (components and related operation) reviewed was intentionally selective so a vertical slice assessment could be performed.

   With respect to plant configuration and performance, this self-assessment found that for the attributes reviewed, the System, Structure, and Component (SSC) configuration and performance were consistent with the design basis as delineated in the DBDs. Attachment B discusses results of this self-assessment as it relates to procedures.

3. Outage Modification Audits (OMAs) - The objective of an OMA is to assure that plant modifications are properly designed, reviewed, approved, and installed, and that new plant configuration is operated in a safe and reliable manner. These audits are conducted periodically using a performance-based methodology consisting of (1) selecting 6-8 modifications, both QA and non-QA scope, to PBNP SSCs that are scheduled to be done during a unit's outage, (2) performing a detailed review of each modification to verify the validity of the design inputs and assumptions including 50.59 safety evaluations, procurement documentation and practices, installation procedures, and close-out of the mod process, and (3) observing various aspects of the installation practices. This is done to ascertain conformance of the design, equipment isolation, housekeeping, and post-modification testing to process and procedural requirements. The purpose of an OMA is to provide a comprehensive look at how the design modification process is being implemented from beginning to end.

   Four OMAs have been completed since their inception in 1988 (and an OMA is currently in progress). These audits found the design and installation activities observed during the audits were generally in compliance with the established procedures. Although these audits continue to find isolated implementation weaknesses, there have been fewer programmatic weaknesses and more programmatic strengths identified in the recently completed audits compared to earlier audits.

4. Programmatic Audits - The Design Engineering functional area is comprised of the following processes:

   - Modifications
   - Design Control
   - Temporary Modifications
   - Design Calculations
   - Engineering Specifications
   - Control/Review of Vendor Documents
   - Engineering Change Requests
   - Q-List Maintenance (classification engineering)
   - Engineering Work Requests
   - Environmental Qualification (EQ)
   - Seismic Qualification
   - Fire Protection Program
   - Probabilistic Safety Analysis

These audits are scheduled to cover all safety-related functions within a two year period. The performance of each functional area is monitored on a continuous basis using available data sources. The resulting performance trends are then utilized to ensure adequate audit coverage of processes and commitments within each functional area.

The Design Engineering functional area has been audited three times in the last five years. These audits have generally concluded that design controls -- including those to maintain SSC configuration and performance consistency with the design bases -- are adequate and are being implemented effectively.

AS-BUILT DRAWING UPGRADE PROJECT

The scope of the PBNP As-Built Drawing Upgrade Project includes: (1) the scanning and redraw of existing drawings for electronic processing; (2) plant walkdowns to verify the accuracy of the drawings relative to actual as-built conditions; and (3) engineering evaluations to determine the consistency of information among related and interfacing drawings. The project was begun with a pilot program in 1990 and is expected to be completed by the end of 1999. WE is including major drawing groups which are vital to plant operations and safety. The key efforts in the as-built drawing upgrade project are the engineering consistency cross-check of the initial drawings, engineering evaluations during the field walkdown for each drawing, and the engineering analyses needed to resolve discrepancies between the drawings and as found condition of the plant. This process also includes canceling existing drawings where information is duplicated or replaced by new drawings.

Discrepancies noted during as-built field walkdowns of a system / panel are promptly reported to the responsible group and/or Operations (if the condition affects a system in service) for an evaluation on any impact on the operability of the system. A Condition Report or Work Order is written to evaluate and correct the condition. Discrepancies noted after the walkdown

(during the data review and drawing upgrade process), are resolved, if possible, using additional field data and analysis of the conflicting drawing information relative to the functional design. If this does not resolve the discrepancy, a Condition Report is written for further engineering evaluation and operability/reportability screenings. Results of post-walkdown return-to-service testing are required to be accepted by Operations.

To date, over 40 unit specific and common systems (which include numerous components and thousands of cables, conductors, and terminations) have been inspected in connection with the As-Built Drawing Upgrade Project. Most of the discrepancies found by these inspections involve the material condition of equipment internals relative to improvements in cleanliness, condition of wiring, wire routing, and terminations. A limited number of non-safety-related components have been found to be mis-wired, inoperable, mis-labeled, or not correctly functioning. The only condition specifically identified by the As-Built Drawing Upgrade Project where safety-related equipment configuration or operation was outside its design bases involved the mixed routing of train-specific wiring to safety-related components in the main control boards. This condition has been reported to the NRC and an operability determination has been made. Modifications are currently in progress to correct this condition. This isolated instance compared to the large number of systems and components reviewed by the As-Built Drawing Upgrade Project suggests that this is not indicative of a generic problem with design basis conformance outside the main control boards.

INSTRUMENT SETPOINT VERIFICATION

PBNP initiated an Instrument Setpoint Verification Program in 1993 to formally calculate the minimum required margins between safety-related instrument setpoints, their analytical limits, and Tech Spec values, taking into account the appropriate instrumentation uncertainties and process measurement effects. ISA Standard 67.04-1994 was used as guidance for the setpoint methodology developed for these setpoint calculations. Many of the analytical limits used in these calculations were researched and documented in conjunction with the PBNP Design Basis Program.

To date, setpoint margins associated with fifteen different reactor trip, safeguards actuation, and other Tech Spec variables have been calculated and all were found to contain sufficient margins from their associated analytical limits to the existing field and Tech Spec settings. Therefore, no setpoint changes have been required. The PBNP Instrument Setpoint Verification program was independently reviewed during a 1994 WE Instrument & Control Audit performed in lieu of an NRC Systems Based Instrument and Control Inspection (SBICI). This audit found both strengths and weaknesses associated with the program. A follow-up NRC inspection noted that the audit observations relating to the Instrument Setpoint Verification Program were being adequately addressed.

TESTING AND SURVEILLANCE PROGRAMS

Testing and surveillance programs also provide reasonable assurance that PBNP SSC configuration and performance are not degraded below acceptable margins. A weak link

between the design bases and the inservice testing acceptance criteria has been identified as a deficiency in DBD Validations, WE assessments, and 1996 NRC inspections. Accordingly, commitments were made to the NRC in September 1996 involving several short-term and long-term corrective actions which affect testing procedures.

The Inservice Test series of procedures implement pump and valve testing requirements of ASME Section XI. The problem identified was that while the Inservice Test procedures comply with Section XI limits for establishing acceptance criteria, the acceptance criteria did not, in all cases, represent design basis limits for the particular component. Accordingly, the inservice testing acceptance criteria for pump and valve testing have been reviewed against their design bases.

The review included comparison of design bases to inservice testing acceptance criteria and to actual component test data. The test data reviewed have demonstrated that both pump and valve components have performed within their design bases. While some test procedures and acceptance criteria have been modified as a result of this effort, this was done so that the test procedures and the acceptance criteria would properly reflect design basis criteria (although the test acceptance criteria may, in some cases, actually be more conservative than what the design bases require). Based on the review results, historical pump and valve performance has conformed to the design bases, and the inservice testing program acceptance criteria, as modified, will continue to conservatively conform to the plant design bases.

A review of other safety-related equipment, including heat exchangers, is currently underway to ensure that inservice testing acceptance criteria reflect design basis requirements. Where acceptance criteria in existing surveillance and testing procedures are not adequately linked to the design bases, the procedure will be revised. A periodic callup and procedure is also being implemented to establish an annual review of the inservice testing program against plant design basis changes. Together with the 50.59 process, these enhancements will establish additional assurance that Inservice Test procedure acceptance criteria are conservative with respect to the design bases.

ENVIRONMENTAL AND SEISMIC QUALIFICATION REVIEWS

PBNP's Environmental Qualification (EQ) Program is designed to ensure that electrical equipment important to safety will function during design basis events which could result in environments that significantly deviate from the normal operating conditions for the equipment (i.e. Loss of Coolant Accident and High Energy Line Break). The program consists of a master list of equipment required to be environmentally qualified, summary sheets identifying the conditions under which the equipment may need to operate and the conditions to which it is qualified, and maintenance requirement sheets identifying specific maintenance, repair, and installation requirements necessary to preserve the environmental qualification of equipment. Additional documents supporting the environmental

qualification of equipment, (e.g., test reports, evaluations, and calculations) are maintained in an indexed filing system. An 8/30/84 NRC Safety Evaluated Report concluded that PBNP's EQ Program was in compliance with the requirements of 10 CFR 50.49. The program has also undergone additional inspections/audits with the most recent audit completed in January 1995. This QA audit (audit report number A-P-94-21) concluded that PBNP's EQ program is sound and is being effectively administered.

Several programs also evaluate the seismic adequacy of PBNP's electrical and mechanical equipment. Most recently, in response to NRC Generic Letter 87-02, PBNP completed and forwarded results of our Unresolved Safety Issue A-46 evaluation on the seismic adequacy of mechanical and electrical equipment. This report, submitted to the NRC as WE letter VPNPD-95-055, and revised by letter VPNPD-96-003, provides the final documentation of the seismic adequacy evaluations for equipment required for safe shutdown of the reactors during and following a seismic event. Other regulatory issues such as NRC IEB 79-14 (piping and piping supports), IEB 79-02 (concrete expansion bolts) and IEB 80-11 (concrete block walls) have resulted in additional evaluations of the seismic adequacy of PBNP SSCs.

These EQ and Seismic qualification programs provide as-built information and documentation that can be utilized as needed to make determinations and evaluations about possible non-conforming conditions that might arise in the future. Additionally, as noted in Attachment F of this letter, PBNP intends to write and validate "Equipment Qualification" and "Seismic Design and Analysis" Design Basis Documents. These DBDs and associated validations will provide additional documentation and assess how PBNP's SSCs are configured to satisfy their environmental qualification and seismic design basis requirements.

QA CLASSIFICATION PROCESS / Q-LIST

The objective of the PBNP Quality Assurance classification process is to provide an accurate and up-to-date QA classification database. The Quality Assurance classification process is comprised of the various procedures and documents used to designate PBNP SSCs with respect to the following classifications: QA scope, safety-related, seismic, EQ, fire protection, and safe shutdown. The process provides the basis for determining and documenting the QA classification (scoping) of SSC and parts in accordance with their quality and functional classification as delineated by QA codes described in the Quality Assurance program.

The classification process is based on system function and operating mode requirements, and the design and licensing bases as described in the FSAR, Technical Specifications, Design Basis Documents, and Fire Protection Evaluation Report. Verification programs for environmental/seismic qualification and Greenline Classification Diagrams, which are color-coded system drawings illustrating the QA classification of components, also support the classification process. The SSC classifications are listed in the PBNP Q-List database controlled within the Computerized History and Maintenance Planning System

(CHAMPS). An accurate and up-to-date QA classification database provides design, quality, procurement, operations, maintenance, and system engineering personnel with a higher level of confidence in the classification of SSC at PBNP, which contributes to a more effective management of the plant configuration.

EXTERNAL ASSESSMENTS

Many external assessments, such as NRC inspections, evaluated issues pertaining to design bases implementation. For example, the objective of Safety System Functional Inspections (SSFIs) is to verify the functionality of selected safety systems by inspecting the performance related attributes of the system with a focus on the body of plant-specific design basis information. Several weaknesses identified by the pre-1990 NRC inspections (EDSFI for example) were related to a lack of available design and engineering information. The EDSFI was conducted prior to the establishment of our DBD program, which, as described earlier in this letter, was designed to capture and document this information. These inspections have also identified weaknesses relating to the lack of a strong link between the design bases and inservice testing acceptance criteria. Without this link, an SSC's performance could be allowed to degrade below its design basis requirements. Similar weaknesses have also been identified by DBD Validation and WE Assessments. As discussed previously in this attachment under "Testing and Surveillance Programs", a review of inservice testing results have indicated that the SSC's tested performance has been consistent with the design basis.

Another recent NRC finding questioned whether the Emergency Diesel Generator performance testing adequately reflected design basis conditions. In this instance, the affected diesel generators were declared inoperable until their performance could be re-verified against a test which better reflected design basis conditions. Although we believe this to be an isolated case, we are currently reviewing 20% of the Operations Technical Specification and Operations Refueling Tests for accuracy and compliance with appropriate initial conditions, return to service lineups, properly specified independent verification, acceptance criteria, and technical specification implementation. Should this review identify either generic issues or significant discrepancies which could negatively impact reactor safety, the scope of the review will be expanded.

ATTACHMENT D

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

The PBNP processes directing the identification of problems, problem evaluation, corrective action identification and implementation, operability determination, reportability determination and trending are governed by three distinct programs:

- Work Order Process
- Condition Reporting Process
- WE Assessments

These processes provide reasonable assurance that problems associated with implementation of PBNP design basis information are identified and appropriately reported to the NRC, and that actions are taken to correct the problems and prevent recurrence.

WORK ORDER PROCESS

The Work Order process provides a mechanism for the identification, recording, and tracking of equipment issues. The Work Order process is administratively controlled by procedure NP 8.1.1, "Work Order Processing" which also specifies how work is to be accomplished. Any person in the NPBU may initiate a work order to have work performed on PBNP components, systems, structures, or grounds. Following completion of the Work Order Tag by the initiator, the Work Order is reviewed by the Work Order Review Group for adequacy and completeness. If the problem or defect described meets the criteria established under the Condition Reporting process, a Condition Report is initiated.

CONDITION REPORTING PROCESS

The purpose of the Condition Reporting process is to ensure that events or conditions potentially adverse to quality or which have the potential to adversely affect the safe and efficient operation of PBNP are promptly identified, evaluated, and corrected to prevent recurrence. The Condition Reporting process is administratively controlled by procedure NP 5.3.1, "Condition Reporting System."

Any WE employee (or contractor through their contractor liaison) may initiate a Condition Report. Condition Reports (CRs) may be written for a number of reasons, but typically fall into one of the following categories:

- Equipment Issues (includes, for example, operability concerns, engineered safety features actuation, deficient designs, repetitive failures)

- Material concerns
- Procedure/Manual/Document discrepancies
- Drawing discrepancies
- Procedure violations
- Inadequate reviews/resolution
- Discrepancies with alarms, setpoints, and calibration
- Personnel errors and work practice deficiencies
- Incorrect QA scoping of systems, structures and components
- Unanalyzed conditions
- Radiological events
- Technical Specification violations
- Procurement issues
- Industrial safety concerns
- Rework issues

The person initiating a CR is responsible for completion of the CR form, including a full description of the event or condition; identification of why the event or condition is potentially adverse to NPBU activities; identification of any immediate or interim corrective measures implemented; and identification of any recommended corrective actions. The initiator is then required to have the CR reviewed by an active licensed Senior Reactor Operator (SRO) for an immediate operability and reportability screening. Plant Operations personnel will take any required actions or the appropriate interim actions to address any potential nuclear safety concerns identified under the CR.

An operability determination is made based on the guidance of Generic Letter 91-18 for CRs that identify a potential degradation or nonconformance of safety-related systems, structures, or components (SSC). Guidance for making an operability determination is provided in procedure NP 5.3.7, "Operability Determinations." If the SSC is determined to be inoperable, the appropriate Technical Specification Limiting Condition for Operation is entered and resolution of the degradation is promptly addressed. If the SSC is determined to be operable, long-term resolution of the condition is pursued as described below.

For Condition Reports involving a reportable plant condition or event, the appropriate NRC notification is made in accordance with 10 CFR 50.72, Immediate Notification Requirements for Operating Nuclear Power Plants or 10 CFR 50.73, Licensee Event Report System. The PBNP Duty & Call Superintendent (DCS) Handbook DCS 2.1.1, "Requirements and Guidance for Immediate Notification to NRC/EPA of 'Significant Events' at PBNP" provides guidance to the DSS, DCS, and duty technical advisor for meeting the NRC reporting requirements of 10 CFR 50.72. Procedure NP 5.2.1, "Licensee Event Reports," describes the reporting criteria and provides interpretive information for submitting Licensee Event Reports as required by 10 CFR 50.73.

Condition Reports (CRs) are classified into two categories: (1) Conditions Adverse to Quality, and (2) Significant Conditions Adverse to Quality (SCAQ). CRs initiated during

QAS internal assessments (i.e. audits) have the prefix "Q" in the tracking number to identify them as Quality Condition Reports (QCRs).

The CR and SCAQ CR designation is determined by the Regulatory Services Group which is responsible for management of the overall Condition Reporting System. The process includes: screening of all CRs and QCRs for reportability and operability (in addition to screening by a licensed SRO); processing of the event or condition into the NUTRK database; prioritization, scheduling, and assignment of the evaluation of the condition; root cause determination; corrective action identification and initiation; corrective action verification; and trending and analysis of the database.

QCRs are initiated by the Quality Assurance Section for potential adverse conditions identified during the performance of audits and surveillances. The Quality Assurance Section ensures that prioritization, scheduling, and assignment of the evaluation of the condition; root cause determination; corrective action identification and initiation; and corrective action verification is properly performed for QCRs.

If it is determined that a CR requires additional evaluation prior to implementation of any recommendations or corrective actions, the most qualified group to perform the evaluation is identified and an evaluation request is initiated. SCAQ CRs and QCRs are required to have this initial evaluation completed within 30 calendar days of the initiation date in order to comply with ANSI N18.7-1976 and N45.2.12-1977 requirements. These evaluations may consist of, but are not limited to, a formal Root Cause Evaluation (human performance issue), Incident Investigation, or an engineering analysis. Upon completion of the evaluation, it should be reviewed within 10 working days for completeness and clarity. The purpose of the evaluation process is to determine if additional corrective action(s) should be taken to prevent degradation of the condition, to determine the root cause(s) and contributing factors leading up to the event, and to determine the action(s) required to prevent recurrence.

Corrective actions are identified and initiated to address the identified adverse condition(s) and to prevent recurrence of the event or condition. Interim corrective actions may be implemented upon discovery of the adverse condition and prior to formal evaluation and implementation of a long-term resolution. Action items are entered into the NUTRK database which provides a means for tracking and monitoring of corrective measures by both the group responsible for the action item and the group who initiated the Condition Report. Corrective actions and evaluations are assigned due dates and priorities in accordance with NPBU procedures. NPBU management's expectation is that the priority and/or due date of a specific action item be commensurate with the significance of the event or condition to which it is related. The NUTRK database maintains information identifying the individual (and his/her manager) responsible for the action item, the individual responsible for verification of the action item, due dates, status, and priority changes. The NUTRK system provides automatic notice to the responsible individual of action items coming due and those that have exceeded their due dates.

Completed action items are reviewed for completeness and adequacy by the responsible group head. The Regulatory Services Group verifies and reviews the completed corrective actions prior to final close-out of a CR; the Quality Assurance Section performs a similar review for QCRs.

All CRs and QCRs are assigned "trend codes" by the Regulatory Services Group following the determination of the root cause(s) of the event or identified condition. Specific trend codes are assigned for when the event occurred, which group caused the event, why the event occurred (root cause), what the event involved, and what system/component was affected. These trend codes are utilized to identify adverse performance trends and repetitive problems. The root cause or "why" trend codes are identified through a formal root cause evaluation, incident investigation, engineering analysis or through the investigation performed by the individual responsible for processing the CR or QCR. All root causes are attributed to human performance issues, equipment performance issues, programmatic/process issues, or are determined to not be an adverse condition.

Based upon internal assessments conducted on the corrective actions process and external evaluations by the NRC and INPO, WE recognizes that the Condition Reporting process needs improvement, primarily in the areas of (1) establishing the proper threshold for identifying problems requiring a Condition Report, and (2) the timely evaluation and resolution of identified problems. Interim actions to improve the Condition Reporting process are currently being implemented. These actions include a daily review of condition reports by a cross-section of group managers and other line management personnel to assesses generic implications, urgency of corrective action, and identification of responsible groups. This combined group review has resulted in an increase in upper management and line group participation and ownership of the condition reporting process. A large increase over the last four months in the number of condition reports issued demonstrates that the threshold for identifying problems requiring a Condition Report has been lowered. We are also currently pursuing long-term actions to strengthen our process.

To provide additional assurance that previously identified conditions are being adequately addressed, we are reviewing existing open item lists including those items identified on other internal lists. Degraded equipment operability issues identified during this review will be evaluated through our Condition Reporting process as described above. Additionally, 20% of the Condition Reports closed since January 1, 1995 associated with PSA safety significant systems are also being reviewed to ensure that the conditions were adequately identified and dispositioned.

INTERNAL QA ASSESSMENTS

The purpose of the Internal Assessment program is to verify compliance with, and the effectiveness of, the nuclear Quality Assurance Program. Internal Assessments consist of internal audits, program surveillances, and work monitoring activities that are focused on verifying the effective implementation of the Quality Assurance Program activities in the NPBU.

Internal audits are formal, planned and documented activities performed by qualified personnel in accordance with established procedures or checklists. The focus of an internal audit is to determine if the applicable elements of the Quality Assurance Program have been adequately developed, documented, and effectively implemented in accordance with specified requirements. Internal audits are typically performed by a team of auditors/evaluators.

Program surveillances have a more focused scope and call for an in-depth look to verify compliance and effectiveness with established requirements. Like internal audits, program surveillances are typically performed by a team of auditors/evaluators.

Both internal audits and program surveillances rely upon written plans and checklists that identify the scope or focus of the internal assessment activity. The checklist is utilized as a guide, and only objective evidence or activities are examined. Any conditions considered to be adverse to quality that are identified during the audit/surveillance are reported and documented via a Quality Condition Report (QCR) per the written requirements of the Condition Reporting System (see above). Observations, such as good practices or management issues, made during the internal assessment that are not considered to be adverse to quality are also noted and included in the final report

Work Monitoring Reports (WMRs) are used to document the effectiveness of work activities and to provide a performance-based input into the internal assessment program. WMRs provide a mechanism to identify and track both weaknesses and strengths that are identified during the monitoring process. WMRs are typically prepared by an individual with the results promptly reported, but WMRs can be prepared in support of an internal assessment or program surveillance.

The WMR program provides a mechanism for observing work activities and evaluating this activity against predefined attributes that are based upon the eighteen criteria of 10 CFR 50, Appendix B. If a concern meeting one of the conditions warranting a Condition Report is identified during the performance of any work monitoring, a CR is generated.

Internal assessments are planned and scheduled in order to comply with the Quality Assurance Program requirements. Various related NPBU activities are grouped together under a functional area, such as Regulatory Compliance or Design Engineering. The performance of each functional area is periodically reviewed using all available data

sources, such as Condition Reports, audits and surveillances, WMRs, and NRC inspection reports. These performance trends are then utilized to ensure adequate assessment coverage of all processes and commitments within each functional area. Based upon this functional area performance trending, NPBU activities are identified by QAS to focus upon for monitoring, assessment, and evaluation.

An Internal Assessment two-year rolling schedule, with the first year being detailed and the second tentative, is established and revised on a quarterly basis. Revisions to the schedule are based upon the functional area performance trending results. The scope, type, duration and frequency of the internal assessment for each functional area can be revised depending upon the results of the functional area performance evaluation. Internal assessments are planned such that each 10 CFR 50, Appendix B criterion is audited at least once every two years.

Internal assessments of corrective actions to facility equipment, structures, systems or method of operation of such equipment that affect nuclear safety are performed under the cognizance of the OSRC. Recent internal assessments concluded that the corrective action program is generally effective, root cause determinations are appropriate, and actions were being taken to prevent recurrence. However, a noted weakness was that a considerable number of CRs did not have priorities or due dates assigned which resulted in action items not being addressed and closed out in a timely manner. This weakness is being addressed.

Management from the organization or process being assessed is kept informed as the audit or surveillance progresses and upon its completion. Internal audit and surveillance reports are prepared following completion of each internal assessment. Audit reports shall be issued no later than 30 calendar days after completion of the post-audit conference. Surveillance reports shall be issued within 30 calendar days of final completion.

ATTACHMENT E

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

Wisconsin Electric believes that current processes and programs provide reasonable assurance that PBNP configuration, as well as operation, are consistent with the established design bases. Wisconsin Electric also recognizes that the processes and programs will continue to evolve as a result of user feedback and process and program evaluations. Our intent is to continually strive for excellence in all aspects of plant performance and safety.

Attachments A through D describe the processes and programs currently being used to verify or maintain the configuration of PBNP consistent with the design bases. These attachments also describe recent initiatives to improve these processes and programs. Current programs and processes include the following:

- Design Basis Document Validation
- 10 CFR 50.59 Reviews
- Instrument Setpoint Verification
- As-Built Drawing Program
- Environmental and Seismic Qualification Reviews
- Testing and Surveillances

In addition to these activities, the following recent initiatives are currently being implemented which will further evaluate and enhance the conduct of operations, maintenance, and testing of PBNP with respect to the design basis.

- Inservice Testing Enhancements
- Post-Maintenance Testing Process Improvements and Procedure Development
- Maintenance and I&C Procedure Upgrades
- Operations Procedure Upgrades

The overall effectiveness of our current processes and programs is evaluated through internal and external mechanisms. In responding to this item, Wisconsin Electric has focused on independent activities that are outside of the implicit review and verification checks of processes and programs which are used to maintain and document the design bases. This effort included an evaluation of historical information as well as a recently performed review to indicate trends and the status of our current processes and programs.

INTERNAL MEASURES OF EFFECTIVENESS

Wisconsin Electric's internal indicators of effectiveness are primarily the results of audits and assessments and the identification of design-related nonconformances. Vertical slice audits and assessments of seven systems and a NRC-monitored, Safety System Based

Instrumentation and Control audit have been performed over the last eight years. These assessments included an evaluation of PBNP configuration with respect to the design bases. Most of these efforts had been augmented with outside technical consultants specializing in assessment and inspection activities to provide outside perspectives. The most recent vertical slice audit was performed in the first quarter of 1996. An additional self-assessment was performed in January 1997 which reviewed design and configuration control issues on two systems. The results of these audits and assessments are summarized in Attachments B and C.

Additionally, the Wisconsin Electric Quality Assurance Section has performed audits of the engineering design organization, which included a reviews of the design and configuration control processes. Several of these QA audits utilized vertical slice inspection techniques in evaluating the adequacy of the design modification process, similar to NRC Safety Systems Outage Modification Inspections.

Historically, Wisconsin Electric's internal measures of effectiveness have shown that the plant configuration and operation is generally consistent with the design bases. A review of the trends of assessments, audits and nonconformances would suggest that the number and significance of issues have remained relatively constant. When the results of these efforts are closely analyzed, however, it can be seen that programmatic problems have become fewer and the technical findings have become more sophisticated and complex. These results suggest that the processes and programs have improved and personnel are challenging calculations and assumptions. Overall, we have become more sensitive to design bases issues due to our DBD efforts coupled with the results of WE assessments and NRC inspection activities.

We therefore conclude that our processes and programs continue to become more effective and our threshold for identifying design related nonconformances has decreased. However, since our internal evaluations continue to identify design and configuration issues, we recognize that there is more work to be done in this area.

EXTERNAL MEASURES OF EFFECTIVENESS

Several external indicators are also available to measure the effectiveness of processes and programs. These external indicators include peer evaluations by industry organizations, commercially required inspections and NRC inspections. Evaluations by industry organizations have consisted of INPO/WANO-supported or initiated activities; national and regional QA peer assessments; and other cooperative professional evaluations of utility activities. The results of these industry evaluations provide an industry-wide measure of performance. These results also provide additional indicators of the effectiveness of utility processes and programs with respect to design and configuration control.

Another measure of effectiveness can be gained from commercially required inspections, such as those performed by insurance carriers. While these inspections focus primarily on

equipment and component performance, plant material protection, and the potential for off-site releases, they have also provided insight on design and configuration control. The results of these efforts are a primary indicator of equipment and component operation within the manufacturer design and service parameters, as well as professional society acceptance standards. These activities also provide a relative comparison of performance with the industry.

Past external evaluations by industry groups have revealed only isolated concerns with plant design and configuration processes and programs. While this is a positive indication, the scope of these evaluations, by themselves, do not provide enough information to fully determine the effectiveness of our processes and programs. The results of commercially required inspections, such as the insurance inspections have also been good. PBNP generally ranks high in meeting insurance carriers' standards and inspection findings are usually corrected promptly. Overall, the inspection results have demonstrated that Wisconsin Electric has adequately maintained and tested certain major pieces of equipment and components and operated them within specified component design and service requirements. Wisconsin Electric is notable in that it is one of a very few utilities that has not filed an insurance claim for equipment failures.

Historically, an additional indicator of effectiveness comes from NRC inspection activities. These include NRC regional and headquarters, individual and team inspections. Several of these have focused on design and configuration control issues at PBNP including a Safety System Functional Inspection of the Electrical Distribution System (EDSFI). However, the NRC has not performed many team inspections at PBNP, recognizing our past efforts in performing internal audits such as the NRC monitored Instrument & Control Audit (audit report number A-P-94-01). NRC inspections have recently been more focused on design and configuration controls.

A review of NRC inspections over the four years, including NRC review of WE internally performed assessments, indicates that there have been relatively few identified design and configuration control deficiencies. The recent increase in inspection activity by regionally-based teams has resulted in the identification of additional design and configuration concerns. Some of these concerns had been previously identified by Wisconsin Electric with corrective actions pending.

Based on the above, we recognize that ongoing efforts to enhance our current process and programs as well as address technical issues as they are identified, need to be supplemented with additional process improvements.

OVERALL CONCLUSIONS

Overall conclusions that can be drawn from the review of the effectiveness of our processes and programs include the following:

- Wisconsin Electric has undertaken a voluntary, systematic, and sustained effort to improve the documentation, availability, and application of PBNP design basis information. We believe that this program is effective and has increased our awareness, understanding, and use of design basis information.

- Our design and configuration control programs and processes are generally good but certain weaknesses continue to be identified in WE assessments and NRC inspections. Trends indicate that there are fewer programmatic problems, pointing to the need for enhanced implementation of programs and processes rather than major programmatic changes.

- Improvements are required to strengthen the effectiveness of the 10 CFR 50.59 process utilized at PBNP. Short-term improvements have been implemented, procedures have been revised, and associated training completed. The long-term improvements require conducting a formal process improvement effort for the 50.59 process.

- We believe that our internal assessment programs, most notably our Vertical Slice Audits, have been effective in providing reasonable assurance that the plant current configuration and performance are consistent with the design and licensing bases.

- Recently identified instances where procedures were found to be deficient may indicate weaknesses in configuration management of procedures relative to design bases. As discussed in Attachment B, several recent initiatives have been completed and others are ongoing in an effort to strengthen the link between design basis requirements and the operation, maintenance, and testing procedures. In addition to these actions, Wisconsin Electric intends to complete the development and implementation of the NPBU Configuration Management (CM) Program Plan. The objective of the NPBU CM Program is to provide an integrated process for ensuring that the PBNP physical plant, the design and licensing basis requirements, and their documentation are synchronized.

- We believe that our Condition Reporting process is effective in identifying and correcting design bases nonconformances. However, we recognize that our Condition Reporting process needs improvement in the areas of (1) establishing the proper threshold for identifying problems requiring a Condition Report, and (2) the timely evaluation and resolution of identified problems. A large increase over the last four months in the number of condition reports issued demonstrates that the threshold for identifying problems requiring a Condition Report has already been lowered.

DESIGN BASIS DOCUMENT STATUS

DBD #	DBD Title	Status	Completion Date	Validation	Internal Assessment	NRC Inspection	Risk Significant System? (1)
1	Auxiliary Feedwater	Complete	1994	Yes	A-P-90-12		Yes
3	Condensate & Feedwater	Complete	1994	Yes			
5	Fuel Handling	Complete	1994	Yes			
6	Instrument and Service Air	Complete	1995	Yes			Yes (Inst. Air)
9	Reactor Coolant System	Complete	1995	Yes	S-A-97-01		Yes
12	Service Water	Complete	1994	Yes	A-P-93-01	SWSOPI, IR 93-012	Yes
17	Vital 120 VAC	Complete	1995	Yes		EDSFI, IR 90-201	Yes
19	Vital 125 VDC	Complete	1994, Rev 1996	Yes	S-A-97-01	EDSFI, IR 90-201	Yes
20	345 KVAC	Complete	1994	Yes			Yes
21	480 VAC	Complete	1994	Yes		EDSFI, IR 90-201	Yes
22	4160 VAC	Complete	1994, Rev 1997	Yes		EDSFI, IR 90-201	Yes
27	Reactor Protection (including elements of the Nuclear Instrumentation system)	Complete	1994	Yes	A-P-91-10		Yes
30	Containment HVAC	Complete	1996	Yes			Yes
31	Control Room HVAC	Complete	1995	Yes			Yes
33	Containment Structures and Penetrations	Complete	1995	Yes	A-P-89-12		Yes
T-35	Accident Analysis Modules (modules 1 to 16)	Complete	1995	No			
T-36	Overcurrent Coordination and Protection	Complete	1996	Yes			
T-44	Post Accident Monitoring (R.G. 1.97)	Complete	1994	Yes		RG 1.97 Special Safety Insp. 5/91	
P-50	Electrical & Mechanical Separation Position Paper	Complete	1996	No			
18	13.8 KVAC (including elements of Gas Turbine system)	Complete	1997	Yes			Yes
24	ESF (Safeguards) Actuation	Draft	1997	Yes			Yes
10	Residual Heat Removal	Draft	1997	In Progress	A-P-88-10		Yes
29	Control Building & Auxiliary Building HVAC	Draft	1997	In Progress		EDSFI, IR 90-201	Yes (2)
4	Chemical & Volume Control	Draft	1997	No	A-P-96-02		
2	Component Cooling Water	Draft	1997	In Progress			Yes
T-40	Fire Protection / Appendix R	Draft	1997	Yes	A-P-93-02		
T-41	Hazards	Draft	1997	Yes			
T-38	Containment Isolation	In Progress	1997	Not Started			
T-39	Equipment Qualification	Not Started	1997	Not Started	Jan-95		
11	Safety Injection (3)	In Progress	1998	Not Started			Yes
T-47	Design Basis Event Combinations	Not Started	1998	Not Started			
13	Spent Fuel Cooling and Filtration	Not Started	1998	Not Started			
T-46	Station Blackout (including elements of Gas Turbine)	Not Started	1998	Not Started			Yes
T-37	Cables & Raceways	Not Started	1998	Not Started			
7	Main Steam & Steam Dump	Not Started	1998	Not Started			Yes
16	Diesel Generator, Fuel Oil & Starting Air	Not Started	1998	Not Started	A-SP-88-02, Also project audits/surv.	EDSFI, IR 90-201	Yes
26	Radiation Monitoring & Protection	Not Started	1998	Not Started			
T-45	Seismic Design & Analysis	Not Started	1999	Not Started			
8	Post Accident Sampling	Not Started	1999	Not Started			
T-42	Instrument Setpoint Calculation Basis	Not Started	1999	Not Started			
25	NSSS Controls	Not Started	1999	Not Started			
14	Turbine Overspeed & Crossover Steam	Not Started	1999	Not Started			

SUMMARY
Total DBDs = 41
System DBDs = 29
Topical DBDs = 12
DBD Position Paper = 1

DBD STATUS
Completed = 19
In Progress = 9
Not Started = 13

Notes
1 - All Maintenance Rule Risk Significant systems (28 total) are addressed in DBDs (except mechanical aspects of the Gas Turbine and its support systems)
2 - DBD-29 includes the risk significant Diesel Generator, Cable Spreading, and PAB Battery & Inverter rooms HVAC systems
3 - Pre-Operating License Safety Injection DBD has been prepared by Westinghouse

ATTACHMENT G - LIST OF ACRONYMS USED

ANSI	American National Standards Institute
AOP	Abnormal Operating Procedures
ASME	American Society of Mechanical Engineers
CHAMPS	Computerized History and Maintenance Planning System
CM	Configuration Management
CR	Condition Reports
DBD	Design Basis Document
DCS	Duty & Call Superintendent
DDP	Design Document Program
DSS	Duty Shift Superintendent
EOP	Emergency Operating Procedure
EPA	Environmental Protection Agency
EQ	Environmental Qualification
FCR	FSAR Change Request
FSAR	Final Safety Analysis Report
I&C	Instrumentation and Controls
ICP	I&C Procedures
INPO	Institute for Nuclear Plant Operations
MI	Maintenance Instructions
MSS	Manager's Supervisory Staff
MWP	Maintenance Work Plans
NPBU	Nuclear Power Business Unit
NSSS	Nuclear Steam System Supplier
NUMARC	Nuclear Management and Resources Council
NUTRK	Nuclear Tracking System
OMA	Outage Management Audits
OSRC	Off-Site Review Committee
PBNP	Point Beach Nuclear Plant
PMT	Post-Maintenance Testing
PUP	Procedure Upgrade Project
QA	Quality Assurance
QAS	Quality Assurance Section
QCR	Quality Condition Reports
RMP	Routine Maintenance Procedures
SCAQ	Significant Conditions Adverse to Quality
SRO	Senior Reactor Operator
SSC	Systems, Structures, and Components
SSFI	Safety System Functional Inspection
Tech Specs	Technical Specifications
USQ	Unreviewed Safety Question
VSA	Vertical Slice Audits
WANO	World Association of Nuclear Operators
WCAP	Westinghouse Commercial Atomic Power
WE	Wisconsin Electric
WMR	Work Monitoring Reports