Consumers Power

POWERING MICHIGAN'S PROGRESS

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U S Nuclear Regulatory Commission
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DOCKET 50-155 - LICENSE DPR-6- BIG ROCK POINT PLANT RESPONSE TO NRC LETTER: REQUEST FOR INFORMATION PURSUANT TO 10 CFR 50.54(f) REGARDING ADEQUACY AND AVAILABILITY OF DESIGN BASES INFORMATION

On October 9, 1996, the U.S. Nuclear Regulatory Commission (NRC) issued a letter, titled "Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information". This letter requests each licensee to submit information that will provide the NRC confidence that each plant is operated and maintained within its Design Bases as defined in 10 CFR 50.2, and that any deviations are reconciled in a timely manner. The NRC request lists five (5) items for which information is to be provided. Licensees were also requested to indicate if design review or reconstitution programs have been undertaken. This letter provides Consumer Power Company's response with respect to the Big Rock Point Plant.

The Big Rock Point Plant, a sixty-seven (67) MWe and two hundred and forty (240) MWt facility, located in Charlevoix, Michigan has been operating for thirty-five (35) years at a seventy-six percent (76%) availability factor. Big Rock Point received its Provisional Operating License on August 30, 1962 and began commercial operation in 1963. The plant will be permanently shutdown in May of 2000.

Consumers Power Company believes that there is reasonable assurance that the Big Rock Point Plant is and has been operated and maintained in accordance with its design basis. The reasons for this belief are summarized below and are more fully described in the attachments.

__Design Basis Verification and Documentation__: The Big Rock Point design bases are described in the Updated Final Hazards Summary Report (UFHSR - Big Rock Point's FSAR) and the Plant Technical Specifications (TS). Big Rock Point was one of the eleven plants that participated in the NRC Systematic Evaluation Program (SEP), which commenced in February of 1977. This

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effort was initiated to review the designs of older operating plants to reconfirm and document their safety. Along with other plants licensed prior to the development of the General Design Criteria, Big Rock Point undertook a comprehensive review of its design against these newer requirements. Big Rock Point also proposed and included with the SEP effort resolutions for the Three Mile Island (TMI) NRC Action Plan, the Unresolved Safety Issues (USIs) program and the Level III Probabilistic Risk Assessment (PRA). These intensive reviews took place during the years 1977 to 1984 and they involved the Consumers Power Company, several expert consultants, the NRC, the Advisory Committee on Reactor Safeguards (ACRS), the Oak Ridge National Laboratory (ORNL), the Lawrence Livermore National Laboratory (LLNL), EG&G and the Franklin Research Center. Over 93 Safety Evaluation Reports (SER's) were issued by the NRC relative to Big Rock Point during this period. The SEP review was designed to provide:

- An assessment of the significant differences between the then current technical positions on safety issues and those that existed when Big Rock Point was licensed,

- A basis for deciding how these differences should be resolved with an integrated plant safety review process (PRA), and

- A documented evaluation of plant safety.

This effort, culminating with the issuance by the NRC of NUREG-0828 in May of 1984, concluded the following with respect to the eighty-five (85) topic areas examined:

- Fifty-three (53) topics met current criteria or were acceptable on another defined basis,

- Two (2) were acceptable based on modifications made during the review, and

- The remaining thirty (30) differed from current criteria in certain aspects and needed the following actions completed to be in effective compliance with all current regulations.

    1. Eighteen (18) Modifications
    2. Twenty (20) Technical Specification/Procedure Changes and/or Development
    3. Twenty (20) Additional Engineering Evaluations

The overall SEP effort has recently been closed by the NRC in a letter dated April 23, 1996. The NRC staff completed its review and resolved the SEP topics which required additional engineering evaluations and/or continuation of ongoing evaluations subsequent to the issuance of NUREG-0828 for Big Rock Point.

Of significance, was the creation of the UFHSR, which the NRC required following the SEP review. The report effort was the product of an extensive review of the integration of plant design and specific licensing requirements and it produced the documentation of the design basis for the plant. This document was submitted on December 22, 1989, with updates following at an annual frequency. The UFHSR is used in the review of design changes, procedure preparation/updates or other activities which need to be reviewed against the design basis.

Thus, although originally licensed thirty-five (35) years ago Big Rock Point has, in effect, through a number of initiatives reconciled its design bases and is maintaining it through UFHSR annual updates. These efforts along with the processes outlined below are believed to have obviated the need for any other design bases reconstitution efforts. The SEP reconciliation resolved major design issues, however, not all design data was reevaluated or recreated. When not available the design basis information is recreated and when appropriate submitted to the Staff for review.

Design Process Controls: Big Rock Point maintains a comprehensive process for assuring that design changes are fully reviewed prior to implementation. Further, as part of this administrative process a Plant Review Committee (PRC), comprised of Senior Site Managers with significant plant experience, reviews proposed changes that are important to nuclear safety. As also described in the attachment, Big Rock Point has developed the Quality Assurance Requirements Matrix (QARM) to provide a means for tracking and assuring in an integrated manner that regulatory commitments are implemented by plant procedures.

In addition, Consumers Power Company formally committed in 1983 to conduct a comprehensive Integrated Assessment (IA) of all open issues (both regulatory and non-regulatory initiated) for Big Rock Point and to develop a living schedule for resolution of important issues. The IA was initially implemented during the performance of the SEP and continues to this day. The purpose of the IA is to prioritize the responses to various design and regulatory topics or initiatives (whether of NRC, industry or plant origin) based on considerations of risk or increase in plant availability attributable to their disposition. The process provides an ongoing mechanism, in addition to the normal configuration and design control processes, to assure that important questions are addressed in a manner commensurate with their safety significance, taking into account the full spectrum of actions already conducted, underway or planned.

The combination of the Integrated Assessment and the PRC review, as well as the other measures for design documentation and control, provide assurance that the plant design bases and the implications for plant design considerations are evaluated as part of the review of prospective design changes and/or initiatives.

Corrective Action Process Controls: The Big Rock Point Corrective Action Program (CAP) provides the controls for identification, evaluation and correction of design discrepancies. This program is used for resolution of site issues including design deficiencies, equipment failures and human performance problems. This program has been recognized by the NRC for its low threshold for problem identification, encouraging a self-assessment culture and the high level of management involvement throughout the root cause evaluation and problem correction processes. The high level of management attention to this process has been noted as a strength by the Institute of Nuclear Power Operations (INPO). As a result of this culture, conservative reportability and operability decisions are appropriately made by plant personnel and supported by plant management. Plant management takes very seriously the maintenance of a safety culture that encourages the identification and resolution of potential problems and has enlisted industry experts (FPI, International) in the training of all site personnel in human error prevention and the use of the CAP to identify and correct deficiencies.

Conservative Design Features/Few Plant Modifications: Another reason for our confidence in the continued operation of Big Rock Point within its design basis, is that the design is inherently conservative and robust. Even as NRC requirements have evolved, only a few major modifications have occurred such as, the redundant core spray system, the reactor depressurization system and the alternate shutdown building. These are examples of additions of new safety systems or significant modifications of existing safety systems which required prior NRC approval. No major modifications requiring prior NRC approval have occurred since 1990. Thus, while Consumers Power Company has carefully evaluated a number of new NRC requirements, these evaluations have demonstrated that in many instances design changes were unnecessary for Big Rock Point (e.g., Station Blackout, Alternate Rod Injection) even though other facilities may have found plant modifications to be necessary. These reviews also provided additional opportunities to further confirm particular elements of the adequacy of plant design.

Commitment for Continued Improvement: A self-assessment undertaken as part of the preparation of this response identified some discrepancies between the plant procedures and the UFHSR. Although these discrepancies did not affect system operability or any fundamental design bases assumptions, a more in depth evaluation of the UFHSR against plant procedures will be conducted to further ensure procedural conformance with the UFHSR. This effort will be completed, including any changes, for the six (6) most risk significant systems by December 31, 1997, and for the rest of the plant systems by December 31, 1998.

Conclusion: Consumers Power Company believes that we understand our design bases and have appropriately documented it in our Plant Technical Specifications and Updated Final Hazards Summary Report. We believe that we have also effectively operated and maintained our Big Rock Point Plant in conformance with the plant design basis. The planned review of plant procedures against UFHSR will further verify these beliefs.

Robert A. Fenech

Robert A.Fenech
Chief Nuclear Officer

CC Administrator, Region III, USNRC
Director, NRR, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Big Rock Point

Attachment 1 — Responses to Questions (a) through (e) of the NRC Letter dated October 9, 1996.

Attachment 2 — Historical Assessment of Big Rock Point Plant Design, Modifications, Licensing Initiatives and Programs.

CONSUMERS POWER COMPANY

Big Rock Point Plant
Docket 50-155 License DPR-06

Response to NRC Letter Request for Information
Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases
Information dated October 9, 1996

At the request of the Commission and pursuant to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits our response to NRC letter dated October 9, 1996 entitled, Response to NRC Letter Request for Information Pursuant to 10 CFR 50.54(f) Regarding Adequacy and Availability of Design Bases Information. Consumers Power Company's response is dated February 6, 1997.

CONSUMERS POWER COMPANY

To the best of my knowledge, information and believe, the contents of this submittal are truthful and complete.

By Robert A. Fenech
Robert A Fenech
VP-Nuclear Operations Department

Sworn and subscribed to before me this 6th day of February, 1997.

Mary Ann Engle
Mary Ann Engle, Notary Public
Berrien County, Michigan acting in VanBuren County, Michigan

My commission expires February 16, 2000.

(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;

SUMMARY

The Big Rock Point processes controlling the design and configuration management functions are reflected in controlled plant procedures. Those procedures establish a coordinated system whereby responsibilities and functions related to each component of design and configuration management are delineated.

Design and Configuration Control Processes

The particular controls related to design change management are, generally, defined in administrative procedures governing a broad spectrum of potential design-related plant changes. Among other matters, those change controls apply to the facility, design specifications, set points, jumper, link and bypass (temporary modifications), functional equivalent substitutions and minor alterations.

In addition, similarly detailed controls are established in administrative procedures for numerous configuration control processes, including procedures; surveillance, testing and inspection; computer software; the conduct of infrequently performed tests or evolutions; work instructions; maintenance; special site testing or processes; safety-related MOV's; drawings; inservice inspection and testing; materials; and procurement.

Finally, to provide added assurance that the evaluation of changes to plant procedures will fully capture potential impacts on other relevant procedural and quality requirements, Big Rock Point has established a Quality Assurance Requirements Matrix which serves as a cross-reference between committed quality requirements and corresponding plant administrative procedure requirements. That matrix is reviewed as part of the procedural modification process.

Safety Evaluations (SE) and Updated Final Hazards Summary Report (UFHSR) Updates

Further, specific administrative procedure controls are defined for performing safety evaluations pursuant to 10 CFR 50.59 for a broad range of changes to the plant or procedures. As part of these evaluations, a determination as to the need to modify the UFHSR is made, providing assurance that the requirements set forth in 10 CFR 50.71(e) are satisfied.

Quality Assurance

Big Rock Point has established processes for design and configuration management in accordance with the provisions of 10 CFR Part 50, Appendix B. The Quality Program Description for Operational Nuclear Plants, CPC-2A (applicable to both Big Rock Point and Palisades) describes the processes established to implement each element of Appendix B. Further, in addition to the procedural controls, information related to specific initial design requirements is also contained in various design documentation sources.

DISCUSSION

The Big Rock Point processes with respect to each of the above activities is described more fully below.

Design Control Processes Description

The Quality Program Description (CPC-2A) sets forth the general programmatic controls with respect to the implementation of 10 CFR Part 50, Criterion 3, Design Control. Among other activities, those controls apply to modifications to safety-related structures, systems and components. Implementing those controls are a number of site administrative procedures which address several categories of design changes.

At Big Rock Point, design modifications are separated into four (4) different categories; (i) facility changes (FC), (ii) specification changes (SC), (iii) setpoint changes (SPC) and (iv) jumper, link and bypass/ temporary modifications (JLB/TM). Two (2) other processes are employed by Engineering for changes that do not affect plant design or design bases; (i) functional equivalent substitutions (FES) and (ii) minor alterations (MA). These categories were created to address the type of modification (hardware/software); the level of control desired; the level of complexity and the effect on plant operation. A brief description of each control process is provided below.

Facility Changes (FC's)

FC's are planned physical or functional changes in plant design or operation accomplished in accordance with the requirements and limitations of applicable codes, standards, specifications, licenses and predetermined safety restrictions. FCs are modifications that result in a change to the plant design or design bases; or a change in equipment operating mode as specified in plant design or design bases. (Administrative Procedure 3.1.1.1)

Specification Changes (SC's)

SC's are those non-complex upgrades to equipment which change the original specifications but retain the same or similar plant operational characteristics, and do not result in any significant change to operating procedures. (Administrative Procedure 3.1.1.2)

Set Point Changes (SPC's)

SPC's provide a method of identifying, authorizing, assigning and documenting setpoint changes on plant systems and equipment such as alarm setpoints, trip setpoints, relay setpoints, instrument ranges and relief valve settings within plant design boundaries. It does not include the change of settings normally carried out by the Operations Department personnel in the performance of their normal duties, as these operations are covered by other procedures. SPCs may be temporary or permanent changes. (Administrative Procedure 3.1.1.3.)

Jumper, Link and Bypasses/Temporary Modifications (JLB/TM's)

JLB/TM's are those modifications which are temporary, disrupt the normal function of the electrical circuit (jumper, link, or wire removed) or piping system (hose, spool piece, or flange removed), etc, and are performed on equipment which will be in service or operable. (Administrative Procedure 3.1.1.4)

Functional Equivalent Substitutions (FES)

The FES is a process utilized to document, justify and approve the replacement or repair of an installed plant component with a non-identical component or part without affecting its design bases or intended functions. (Administrative Procedure 3.1.1.5)

Minor Alterations (MA's)

MA's are planned physical changes (additions or deletions) to plant systems and facilities that do not meet the definition of a plant design change, do not alter performance specifications of installed plant systems, and are below the threshold of change that are implemented via the FC or equipment SC processes. (Administrative Procedure 3.1.1.8)

As enumerated below, all of the above design modification control processes require preparation of a Safety Evaluation in accordance with Administrative Procedure 1.11. This procedure provides the guidelines for performing Safety Evaluations and establishes the process to address the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments". This procedure also requires a revision to the UFHSR if the item being evaluated is changed from its description in the UFHSR.

An important element of the design change control process at Big Rock Point is the review of proposed design changes by the Plant Review Committee. The PRC is composed of senior plant management representing key engineering, operations, maintenance and radiation protection. The PRC reviews, among other matters, proposed changes or modifications to plant systems or equipment that affect nuclear safety. This review includes the Safety Evaluation that was prepared as discussed above. PRC recommends in writing to the Plant Manager approval or disapproval of proposed

changes or modifications and renders determinations in writing with regard to whether or not the item constitutes an unreviewed safety question. This element of the design change process provides further assurance that full consideration is given to potential implications of proposed design changes to the plant. (Administrative Procedure 1.4)

Configuration Control Processes Description

This section describes those processes which provide assurance that plant procedures, processes and documentation are properly maintained. The processes addressed here include controls of 1) operations, 2) design, 3) maintenance, 4) licensing, 5) training, and 6) procurement. These components are taken from INPO 87-006 Model of Configuration Management. Figure A provides a representation of how the specific Big Rock Point plant procedures are used to provide the components of configuration management. Configuration management is an integrated process whereby:

1. The design requirements for plant structures, systems, components, software and hardware are defined and documented,

2. Changes to design requirements are identified, documented, controlled, evaluated and approved or disapproved,

3. Approved design changes are installed and tested per applicable code,

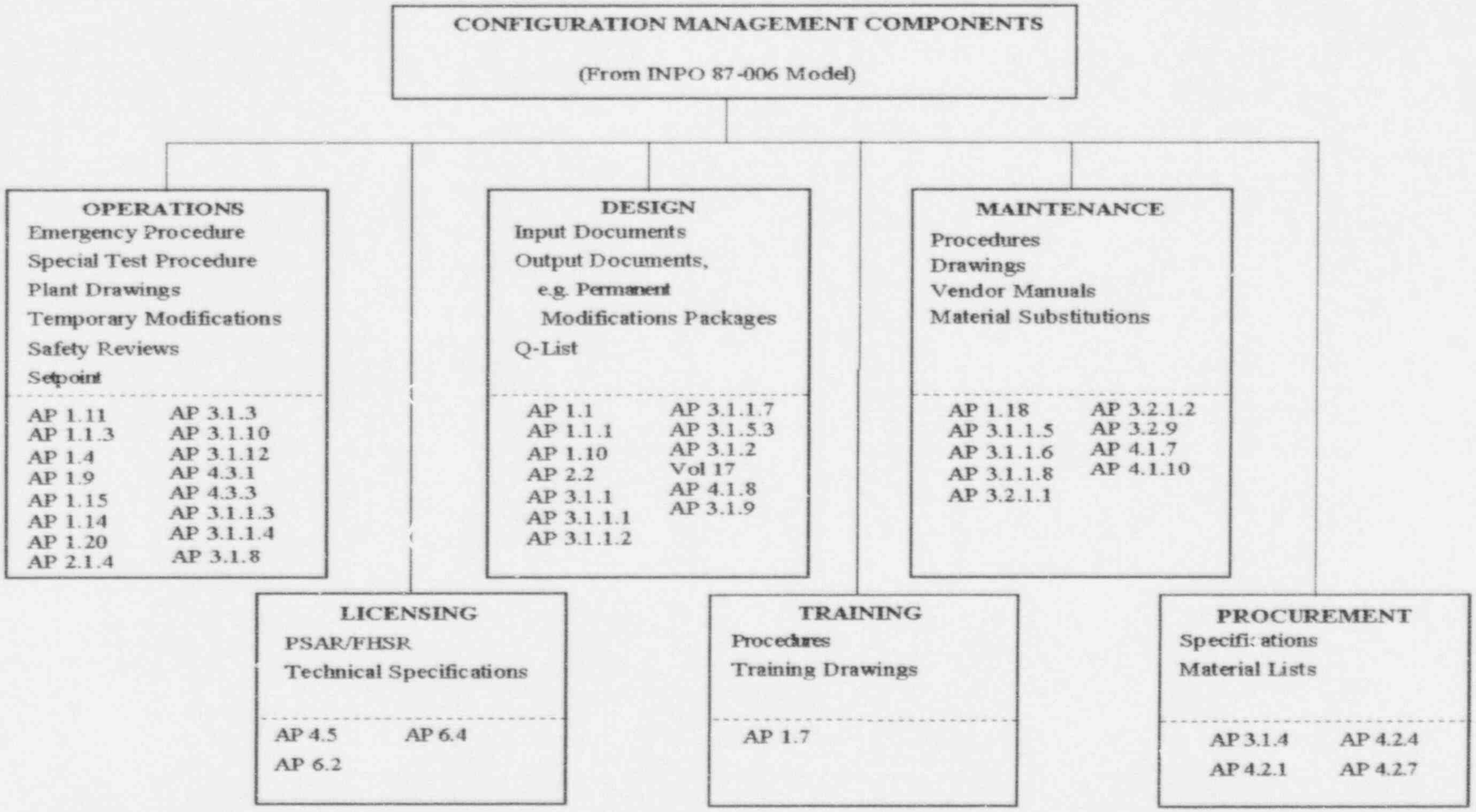
4. Approved design changes and implementation status are recorded and reported through the life of the plant, which results in the accurate implementation of design output information into the physical configuration of the plant (i.e., the "as-built" status matches the design documents), and

5. Plant configuration documents specifying operations, maintenance, testing, installation, procurement and training requirements are updated and maintained consistent with the plant design.

The components of the configuration control processes identified in Figure A reflect the configuration management controls in effect at Big Rock Point. As a general observation, it should be noted that many of these processes are in fact generally controlled by the Quality Program Description (CPC-2A). Implementation of the particular control mechanisms are set forth in the specific administrative procedures referenced below.

ATTACHMENT 1
response (a)

FIGURE A



Operations

EO? Verification and Va'idation, A Iministrative Procedure 1.1.3

This procedure specifies the verification and validation requirements for new or revised Emergency Procedures, (EOP/EIP). Validation is performed to ensure that the Emergency Procedures are logically structured to provide sufficient information for qualified operators to perform their duties in an emergency. All revisions to the Emergency Procedures will require processing in accordance with the procedures program described in Administrative Procedure 1.1.

Plant Review Committee, Administrative Procedure 1.4

This procedure describes the review function of the Plant Review Committee (PRC) and its relationship with the Nuclear Performance Assessment Department (NPAD). Review duties are assigned to individuals of the Nuclear Operations Department and the Plant operating staff as well as applicable committees. In this context, review denotes a deliberately critical examination, including observation of plant operation, procedures and of certain proposed safety actions, and after-the-fact investigations of abnormal conditions. Reviews are conducted as detailed in the Big Rock Point Plant Technical Specifications.

Surveillance, Testing, and Inspection Program, Administrative Procedure 1.9

The purpose of the Surveillance Testing and Inspection Program is to assure that safety related systems and components comply with Technical Specification requirements and applicable codes and standards for testing and inspections.

Safety Evaluations, Administrative Procedure 1.11

This procedure provides guidelines for performing Safety Evaluations and establishes the process to meet the intent of the Title 10, Code of Federal Regulations, Part 50.59 "Changes, Tests, and Experiments," as required by Plant Technical Specifications.

Reactivity Management Program, Administrative Procedure 1.14

The purpose of this procedure is to promote effective reactivity management at Big Rock Point Nuclear Plant and to promote a conservative operating philosophy in which safety and core integrity take precedence over power production and associated activities. It also provides guidance to work activities that have a potential to affect reactivity.

Conduct of Infrequently Performed Tests or Evolutions, Administrative Procedure 1.15

This procedure provides guidance for additional procedure requirements and Management involvement for Infrequently Performed Tests or Evolutions (IPTE). IPTEs may place the plant equipment and operators outside the bounds of normal operating procedures and training. Additional procedure requirements and Management involvement are intended to assure; an environment is established and maintained that places high priority on preserving plant margins of safety; sufficient preparation and support for the test or evolution; Management/supervisory oversight and control during the test or evolution.

Maintenance Rule, Administrative Procedure 1.20

This procedure defines the responsibilities and administrative controls for implementing 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants". This procedure also provides definitions of terms used in regard to the Maintenance Rule and the reporting process associated with periodic assessments.

Plant Status and Equipment Control, Administrative Procedure 2.1.4

This procedure provides specific requirements for reactor trip control, load control, Jumpers, Links and Bypasses/Temporary Modifications, outage requests, Instrument and Control indications, locked valves and drain control. It also provides general operations administrative control requirements for operating the plant.

Jumper, Link and Bypasses/Temporary Modifications, Administrative Procedure 3.1.1.4

JLB/TM's are those modifications which are temporary, disrupt the normal function of the electrical circuit (jumper, link, or wire removed) or piping system (hose, spool piece, or flange removed), etc, and are performed on equipment which will be in service or operable.

Special Site Tests, Administrative Procedure 3.1.3

This procedure describes the administrative control for Special Site Tests (SST). SST include infrequently performed activities that involve; operation of plant systems or components to establish characteristics or values not previously known; or operation of plant systems or components to determine that they function in accordance with predetermined specifications.

Revisions to Plant Drawings, Administrative Procedure 3.1.8

This procedure provides controls for processing new drawings and revisions to Plant drawings.

Operating Experience Review, Administrative Procedure 3.1.10

This procedure establishes the requirements and responsibilities of the Nuclear Operating Experience Review (NOER) Program at Big Rock Point. It applies to the screening, review, evaluation, implementation and disposition of NOER information to provide a mechanism for supplying feedback of operating experience to the plant staff. NOER actions are generally controlled by the Corrective Action Program.

Set Point Changes, Administrative Procedure 3.1.1.3

Set Point Changes provide a method of identifying, authorizing, assigning and documenting setpoint changes on plant systems and equipment such as alarm setpoints, trip setpoints, relay setpoints, instruments ranges and relief valve settings within plant design boundaries. It does not include the change of settings normally carried out by the Operations Department personnel in the performance of their normal duties, as these operations are covered by other procedures. Set Point Changes may be temporary or permanent changes.

System Engineering, Administrative Procedure 3.1.12

This procedure defines the role of the System Engineer in assuring safe and effective plant operations, defines System Engineering responsibilities, and establishes management expectations for the performance of assigned System Engineer duties. This procedure applies only to those system functions that are subject to Maintenance Rule (10 CFR 50.65) requirements. The protection, monitoring and improvement of these Maintenance Rule system functions are the primary focus of the System Engineering Program at Big Rock Point. Plant systems not subject to Maintenance Rule requirements are not assigned a System Engineer.

Shutdown Risk Management, Administrative Procedure 4.3.1

This procedure defines responsibilities and requirements necessary to ensure safe plant operations during shutdown conditions.

Planning and Scheduling, Administrative Procedure 4.3.3

This procedure defines the management organization, departmental responsibilities and processes required to effectively plan, schedule and perform work at Big Rock Point.

Design

Procedures Program, Administrative Procedure 1.1

This procedure describes the general requirements and methods for uniform review, revision and approval of Big Rock Point Plant procedures and/or work instructions. This procedure requires use of the QARM.

Procedure Writer's Requirements and Guidelines, Administrative Procedure 1.1.1

This procedure provides requirements and uniform general guidance for the development and preparation of plant procedures.

Procedure Issuance, Administrative Procedure 4.1.8

This procedure establishes the issue, revision, control and distribution of plant procedures.

Computer Software Control, Administrative Procedure 1.10

This procedure provides guidelines for the control and Quality Assurance requirements for computer software.

Reactor Engineering, Administrative Procedure 2.2

This procedure defines the responsibilities and describes programs and procedures of the Big Rock Point Reactor Engineering Group.

Plant Modifications, Administrative Procedure 3.1.1

This procedure identifies plant modifications processes, Facility Changes (FCs), Specification Changes (SCs), Set Point Changes (SPCs), Jumpers, Links and Bypasses/Temporary Modifications (JLB/TMs) and provides reference to the appropriate plant procedure to be utilized for each specific type of modification.

Facility Changes, Administrative Procedure 3.1.1.1

Facility Changes are planned physical or functional changes in Plant Design or operation accomplished in accordance with the requirements and limitations of applicable codes, standards, specifications, licenses and predetermined safety restrictions. Facility Changes are modifications that result in a change to the Plant Design or Design Basis; or a change in equipment operating mode as specified in Plant Design or Design Basis.

Specification Changes, Administrative Procedure 3.1.1.2

Specification Changes are those non-complex upgrades to equipment which change the original specifications but retain the same or similar plant operational characteristics, and do not result in any significant change to operating procedures (GOPs, SOPs, EOPs, ALPs and ONPs).

Engineering Work Packages, Administrative Procedure 3.1.1.7

This procedure provides direction for the use, preparation, review, approval and revision of Engineering Work Packages generated for the implementation and testing of plant modifications.

Engineering Analyses and Sketches, Administrative Procedure 3.1.2

This procedure provides guidance in the utilization, preparation, review and revision of Engineering Analyses (EA) and Sketches (SK).

Repair Replacement Program, Administrative Procedure 3.1.5.3

The purpose of this procedure is to specify the methods of implementing the ASME Section XI, 1986 Edition Requirements for Repair/Replacement of ASME Class 1, 2 and 3 systems and components. As allowed by Code Case N 389, later editions of this code when referenced in Title 10, Code of Federal Regulations, Part 50.55a may be used for additional guidance and this Case number shall be identified on the NIS-2 form when used.

Equipment Data Base (EQDB), Administrative Procedure 3.1.9

This procedure defines the capabilities of the Advanced Maintenance Management System (AMMS)-EQDB, provides guidance for on-line use and establishes guidelines and defines responsibilities for maintaining accurate data within the EQDB which is essential for an efficient AMMS computer system.

Quality List, Plant Procedures Volume 17

The Q-List (Volume 17) identifies the functional and quality classifications of structures, systems, components, chemicals and consumables. Items identified in the Q-List as safety-related (SR) are subject to the applicable requirements of CPC-2A, Quality Program Description For Operational Nuclear Power Plants. Safety-related items are those items which assure (1) the integrity of the reactor coolant pressure boundary (2) the capability to shut down the reactor and maintain it in a safe shutdown condition and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

Maintenance

Control of Work Instructions, Administrative Procedure 1.18

This procedure establishes requirements for development, review, approval, revision, cancellation and control of Work Instructions used to support maintenance and modification activities at Big Rock Point Nuclear Plant. Work Instructions are intended to support activities where instructions are typically provided in the Maintenance Order job plan and procedures are not required.

Functional Equivalent Substitutions, Administrative Procedure 3.1.1.5

Functional Equivalent Substitution is a process utilized to document, justify and approve the replacement or repair of an installed plant component with a non-identical component or part without affecting its design basis or intended functions.

Safety Related Motor Operated Valves, Administrative Procedure 3.1.1.6

This procedure defines the responsibilities for satisfying the requirements of NRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and outlines the administrative controls in place that provide assurance that the motor-operated valves (MOVs) will function when subjected to the design-basis conditions (both normal and abnormal events) of the plant.

Minor Alterations, Administrative Procedure 3.1.1.8

Minor alterations are planned physical changes (additions or deletions) to plant systems and facilities that do not meet the definition of a plant design change, do not alter performance specifications of installed plant systems, and are below the threshold of change that are implemented via the Facility Change or equipment Specification Change processes.

Performance of Maintenance, Administrative Procedure 3.2.1.1

This procedure establishes the responsibility and administrative controls for all maintenance and modification activities performed by the Plant Maintenance Department. In addition, all other Consumers Power Company personnel and non-plant organizations are expected to comply with the requirements of this procedure.

Post Maintenance Testing, Administrative Procedure 3.2.1.2

This procedure establishes the responsibility and administrative controls necessary to perform post maintenance test activities.

Control of Special Processes, Administrative Procedure 3.2.9

This procedure establishes requirements and responsibilities to ensure that special processes are adequately controlled.

Drawings/Specifications, Administrative Procedure 4.1.7

This procedure describes the method of revision, control and distribution of the Big Rock Point Plant drawings and specifications. It also establishes the method for processing of controlled drawings/specifications.

Vendor Equipment Technical Information Program, Administrative Procedure 4.1.10

This procedure establishes the control, revision, distribution and use of Plant vendor equipment technical information that is used for the performance of activities affecting the quality of safety-related structures, systems or components.

Licensing

Big Rock Point Integrated Plan, Administrative Procedure 4.5

This procedure establishes requirements related to the Big Rock Point Integrated Plan. The purpose of the Integrated Plan is to assess, coordinate, and schedule necessary work at Big Rock Point Plant. Objectives of the Plan include; satisfaction of regulatory requirements, providing sufficient lead time for modifications and other activities, minimizing changes seen by plant operators, effective management of financial and human resources and specification of the framework for changes to developed issue resolutions and associated schedules.

<u>FHSR Management, Administrative Procedure 6.2</u>

This procedure provides the guidance for revising the Updated Final Hazards Summary Report and the subsequent submittal to the Nuclear Regulatory Commission in accordance with 10 CFR 50.71(e).

<u>Commitment Tracking, Administrative Procedure 6.4</u>

This procedure establishes the requirements and responsibilities for the administration of the Big Rock Point Commitment Tracking System. It also addresses the control of incoming and outgoing licensing correspondence and the control, tracking and resolution of any commitments made.

Training

<u>Master Training Plan, Administrative Procedure 1.7</u>

This procedure identifies the responsibilities and requirements for the indoctrination, training and re-qualification training programs necessary for Big Rock Point personnel to assure safe, efficient operation and maintenance of the plant. This procedure has supplementing procedures which specify training requirements for personnel. These requirements include Q-list, safety evaluation and other design bases type training.

Procurement

<u>Preparation & Control of Elec, Civil, Mech Specifications, Administrative Procedure 3.1.4</u>

This procedure establishes criteria and guidelines for the usage, contents, and control of existing civil, electrical, and mechanical engineering specifications and revisions.

<u>Material Control, Administrative Procedure 4.2.1</u>

This procedure establishes requirements for receiving, handling, storing, issuing and returning to stock controlled materials, both safety-related and nonsafety-related.

<u>Procurement - General Requirements, Administrative Procedure 4.2.4</u>

This procedure defines general requirements, responsibilities, and methods for procuring materials and services.

Contracted Services, Administrative Procedure 4.2.7

This procedure establishes the requirements and responsibilities for procuring and controlling contracted services both safety-related and nonsafety-related that will affect Big Rock Point systems, structures, components or operational actions.

Safety Evaluation Process - 10 CFR 50.59

Administrative Procedure 1.11 provides guidelines for performing Safety Evaluations and establishes the process to meet the intent of the Title 10, Code of Federal Regulations, Part 50.59 "Changes, Tests, and Experiments," as required by plant Technical Specifications.

The Big Rock Point process for evaluating new plant activities and changes against 10 CFR 50.59 criteria encompasses a broad range of proposed actions. In that regard, Safety Evaluations are conducted for proposed:

- New procedures,
- Procedure revisions determined to be a "Change-of-Intent",
- Procedure revisions which alter conclusions of an existing Safety Evaluation,
- Procedure cancellation,
- New tests or experiments,
- Revised tests or experiments,
- Modifications to the Facility, (FC, SC, SPC, JLB/TM, MA, FES),
- Engineering Design Changes which alter the conclusions of an Existing Safety Evaluation, and
- Technical Specification Changes.

As an additional level of assurance regarding the adequacy of the safety evaluations performed under this procedure, the Plant Review Committee, described in Administrative Procedure 1.4, conducts a documented review of all new procedures and applicable non-editorial procedure revisions, as required by Technical Specification 6.5.1.6(a), and assesses the adequacy of the proposed unreviewed safety question determination. The PRC also recommends approval (disapproval) of the procedure (or revision) to the Plant Manager. In addition, subsequent to implementation (and no later than the next regularly scheduled PRC monthly meeting), PRC reviews, again as required by Technical Specification 6.5.1.6 (a), all "no change of intent" temporary changes to procedures.

The Nuclear Performance Assessment Department (NPAD) performs a post-review of all unreviewed safety question determinations. NPAD also reviews a sample of the safety screening reviews which have concluded that an unreviewed safety question determination is not required.

UFHSR Control - 10 CFR 50.71(e)

The UFHSR for Big Rock Point was completed in 1989. The update was not intended nor expected to address the descriptions and degree of detail contained in the Standard Review Plan for Modern Plant Summary Analysis Reports. Attachment 2 provides a description of UFHSR contents.

The UFHSR is utilized during preparation of the Safety Evaluation. If the Safety Evaluation determines an item to be changed differs from how it is described in the UFHSR, then the UFHSR will require a change. If the change involves an unreviewed safety question then NRC approval is required. Activities which may impact the UFHSR and which must be reviewed for inclusion or revision to the document include the following:

- Plant Modifications,
- Program or Procedure Changes,
- License or Technical Specification Changes including supporting Accident and Transient Analyses results,
- Exemptions including supporting analyses, and
- Generic Evaluations (SEP Topics, Generic Letters, IE Bulletins).

Administrative Procedure 6.2 provides procedural controls for revising the UFHSR and the subsequent submittal to the NRC in accordance with 10 CFR 50.71(e).

In addition, Safety Evaluations conducted in accordance with Administrative Procedure 1.11 also require a revision to the UFHSR if the item being evaluated is changed from its description in the UFHSR.

In summary, a comprehensive process exists by which Big Rock Point captures proposed changes to the plant, or procedures, for performance of safety evaluations and USQ determinations and translates these changes to the UFHSR.

Quality Assurance - 10 CFR Part 50, Appendix B

The Big Rock Point processes for design and configuration control are described above. As noted above, the Big Rock Point Quality Assurance program elements are set forth generally in CPC-2A, "Quality Program Description for Operational Nuclear Power Plants" and implemented through Administrative and sub-tier procedures. The assurance of adequately maintaining the procedures which implement the Quality Assurance program, and those elements concerning design and configuration control in particular, is provided in large measure by the particular procedural controls identified in Figure A. In fact, Administrative Procedures are maintained in compliance with those ANSI Standards and Regulatory Guides committed to in CPC-2A.

To provide additional assurance that any changes to those procedures adequately capture the fundamental CPC-2A QA requirements, Big Rock Point has established a document titled "Quality Assurance Requirements Matrix (QARM)." That document cross-references QA basis document requirements to the particular steps in the Administrative Procedures that implements the requirement. In particular, the Procedure Sponsor responsibilities include verification that quality assurance requirements are met for Administrative Procedures, using the QARM. In addition, the sponsor is to update the QARM, as appropriate, for changes generated during the procedure revision process.

Applicable elements of CPC-2A are also applied to emergency plans, plant security plans, radiation and fire protection plans as described in those plans. Certain elements of CPC-2A may be selectively applied to nonsafety-related items based on commitments in the plant UFHSR or requirements of the plant Technical Specifications and other docketed analyses. Such items are Q-classified and are included in the plant Q-List because they are important to nuclear safety.

Procedure Temporary Changes

Finally, because of the importance of maintaining formal controls on procedures implementing quality activities, Big Rock Point has provided specific controls related to situations where a temporary change to a procedure may be proposed.

Temporary changes to approved procedures are permitted, except to Administrative Procedures. The purpose of temporary changes is to ensure correct procedural use in the context of temporary system or equipment conditions and not to discourage or circumvent the normal procedure revision process. Temporary changes are approved by a holder of a Senior Reactor Operators License and a PRC member.

Temporary changes to approved procedures are permitted without prior PRC review and approval only when the change does not constitute a change of intent as evidenced by its not resulting in any of the following:

1. Altering the purpose/scope,
2. Altering Technical Specification and UFHSR information,
3. Adding or deleting (altering) acceptance criteria or setpoints,
4. Altering calculations or formula involved in achieving procedure results,
5. Changing the "Q-List" determination (i.e., the safety evaluations),
6. Change the sequence of inspections, tests or other operations important to safety,
7. Change in plant configuration, including temporary modifications, jumper-link-bypass, and setpoint changes, and
8. Altering the response/resolution of a Resident (on-going) commitment made to the NRC.

RESPONSE (A) CONCLUSIONS

The overall processes for design and configuration management controls are described above. These processes are comprehensive and provide reasonable assurance that plant design bases requirements are controlled and translated into plant design and design changes.

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

SUMMARY

Consumers Power Company concludes that plant procedures have adequately incorporated design basis requirements. The Systematic Evaluation Program (SEP) project included a review of plant design relative to the General Design Criteria and of procedures, as necessary to reflect the Big Rock Point design basis. To provide assurance of continued conformance to the SEP recommendations, Consumers Power Company recently conducted a verification of current implementation of the SEP recommendations impacting plant procedures and found those recommendations to have been appropriately maintained in the procedures.

In addition, numerous initiatives subsequent to the SEP have involved further reviews of a broad spectrum of plant/system design basis that have included a number of plant risk significant systems. The Big Rock Point simulator development also entailed an extensive review of design bases parameters and resulted in procedural improvements. These initiatives have found that the plant design bases were adequate and that limited procedural changes could be instituted to enhance plant operation.

To provide further assurance that plant procedures had adequately captured the design basis, a self-assessment was conducted of procedural conformance to the requirements stated in the Updated Final Hazards Summary Report (UFHSR) and the Technical Specifications (TS). This review identified no significant problems or examples of non-conformance with the plant design basis. However, the review did uncover questions related to procedural conformance with elements of the UFHSR. Accordingly, Consumers Power Company concluded that additional reviews are warranted, to ensure procedural conformance with the UFHSR.

The above activities have provided assurance that the plant procedures adequately reflect the plant design basis and that the additional actions planned will provide yet further assurance of procedural conformance to the plant design basis.

DISCUSSION

Big Rock Point's original design basis predates most existing regulatory requirements. Nonetheless, the many initiatives undertaken since receipt of the initial plant license have resulted in updating the original design basis and provided assurance that the plant is operating in accordance with its design basis.

Big Rock Point was designed, constructed, tested and began operation prior to Part 50, Appendix A - General Design Criteria (the Safety Guides that became part of the Regulatory Guide Series, Part 50), Appendix B - Quality Assurance Criteria for Nuclear Power Plants and publication of the Standard Review Plan (SRP). Since initial criticality, extensive AEC/NRC and licensee efforts have been devoted to further defining and documenting the facility's design basis. As part of the original licensing effort, two revisions to the original Final Hazards Summary Report were issued dated March 12 and 19, 1962. During this period General Electric and Bechtel design documents were only supplied to support Staff licensing reviews. These documents were not formerly maintained

The Big Rock Point design bases have evolved since its first operation in 1962. As discussed in Attachment 2, the robust design of Big Rock Point resulted from application of the design standards in effect prior to the current regulations and guidance documents. The Systematic Evaluation Program (SEP) reviewed the plant in the late 1970's and early 1980's in concert with the Three Mile Island (TMI) Action Plan, Big Rock Point Level III PRA results and the Unresolved Safety Issues (USIs) program. This evaluation benchmarked the plant's design against the then current standards and included an assessment of the major modifications that have been performed at the plant. Big Rock Point considers the SEP a significant design basis validation and therefore has not undertaken additional design bases reconstitution programs.

The SEP reviewed plant operational methods and required some procedural changes to implement the design basis as established by NUREG-0828, Integrated Plant Safety Assessment (IPSAR). These procedural requirements are reflected in the Updated Final Hazards Summary Report (UFHSR). No major modifications which have required prior NRC approval have occurred since 1990. Major modifications such as, the redundant core spray system, the reactor depressurization system and the alternate shutdown building are examples of additions of new safety systems or significant modifications of existing safety systems that occurred prior to or during the conduct of the SEP.

With respect to engineering design and configuration control processes, administrative procedures were developed in the mid 1970's to assure that processes are in place to translate design basis requirements into working level procedures. The Big Rock Point Quality Assurance Requirements Matrix (QARM) was later developed and is used to assure that regulatory commitments made in the Quality Assurance Program Description for Operational Nuclear Power Plants (CPC-2A) are translated into administrative procedures and subsequently into operating, maintenance and testing procedures.

Prior to completion of the UFHSR rewrite, a working level procedures upgrade was completed which included verification that Technical Specifications (TS) requirements were correctly translated into the procedures. During the UFHSR rewrite, when any procedural discrepancies were identified corrective actions were taken. However, the self-assessment performed in response to this letter discovered that working level procedures were not in all cases adequately controlled against the UFHSR. The identified discrepancies did not require changes that affect design bases.

Subsequent to the UFHSR rewrite, the development of the Limited Scope Simulator for Big Rock Point verified key design bases parameters and corrected plant procedures when discrepancies were found. This initiative was completed in the early 90's.

As mentioned above, the self-assessment performed as part of this response indicated that some discrepancies do exist between the plant procedures and the UFHSR. Elements of this self-assessment included the following:

- Thirty-one requirements were selected from the Tech Specs for correlation to plant procedures. Applicable sections of the UFHSR were reviewed for the same or similar requirements. Twenty-seven of these requirements were specifically addressed in forty-four plant procedures (including twenty surveillance procedures). The other four requirements have not been specifically incorporated into plant procedures. However, reasonable assurance due to the nature of the requirements exists that they have been met. Procedural enhancements will be made to incorporate the requirements. Because of these identified weaknesses, a formal review of the UFHSR against plant procedures will be performed.

- SEP was the most significant confirmation of the Big Rock Point design basis acceptability. A review of the procedural changes identified by SEP was performed to support the rationale that the plant configuration was being maintained consistent with the design basis. The review confirmed that all (i.e., twenty) of the IPSAR Category 2 items that addressed changes to the Technical Specification and procedures were correctly dispositioned.

Additional evidence that the Big Rock Point design basis have been translated into the plant

procedures is presented by focusing on: projects, response-to-rulemaking, generic letter responses, inspections and corrective actions. With exception of actions taken in the context of the SEP, the following examination concludes that the plant design basis have not been significantly changed regardless of the NRC's issue. Whether the Staff's initiative was a rule, generic letter or inspection, the fact that the design basis did not change underscores Big Rock Point's inherent design margin as well as the success of the SEP in promulgating the design basis into plant procedures.

Examples of the initiatives reviewed, include:

1. Systematic Evaluation Program (SEP),
2. Station Blackout Rule (SBO),
3. Instrument Air Generic Letter (GL) 88-14,
4. Motor Operated Valve Testing - Generic Letter (GL) 89-10,
5. Limited Scope Simulator (LSS),
6. ECCS Design Margin, and
7. Electrical Distribution System Functional Inspection (EDSFI).

Following discussion of these topics, an example of how the Big Rock Point Corrective Action Program (CAP) has supported design basis re-evaluations and how they are promulgated into procedures is presented. In other words, in the resolution of identified corrective actions the design basis is often times revisited in detail. When appropriate, the lessons learned are used to improve procedures. The CAP discussion provides examples of corrective action closure to improve procedures. Subsequent to the CAP examples, the results of a self-assessment performed to support response (b) are presented. This self-assessment concluded that significant deficiencies did not exist between the plant procedures and the UFHSR. Some weaknesses were identified that resulted in our commitment to a formal review of the plant procedures against the UFHSR.

Systematic Evaluation Program (SEP)

The NRC's Systematic Evaluation Program (SEP) which began in 1977, provides substantial assurance of the design basis validation for Big Rock Point. Big Rock Point has not undertaken any subsequent design bases reviews nor reconstitution programs as no major plant modifications which required prior NRC approval have occurred since 1990. The SEP was designed to provide a framework for reviewing the designs of older operating nuclear power plants to reconfirm and document the safety of these facilities. The review provided, 1) an assessment of the significance of differences between the NRC technical positions on safety issues and those that existed when the particular plant was licensed, 2) a basis for making decisions on how these differences should be resolved in an integrated plant review and 3) a documented evaluation of plant safety. This program benchmarked the plant design basis to the review criteria (based on licensing criteria such as regulations, guides and SRP review criteria, or the equivalent of such criteria) in 85 of the 137 applicable "topic" areas. The resolution culminated in a Big Rock Point specific report,

NUREG-0828, the "Integrated Plant Safety Assessment Final Report", dated May 1984 (which also included the resolutions from the Three Mile Island Action Plan, Probabilistic Risk Assessment and Unresolved Safety Issues of the time).

Closure of the SEP effort was documented via Staff letter dated April 23, 1996. The letter provided a cross-reference to identify the documentation that tracked the resolution of the SEP topics that were open at the time of issuance of NUREG-0828.

Big Rock Point instituted an Integrated Assessment (IA) of all open issues (both regulatory and non-regulatory) as part of the SEP. Consumers Power Company formally committed in 1983 to include the IA in the plant license. The IA approach ensured the success of the SEP as a design basis validation effort. The NRC's assessment of these programs was performed by independent contractors such as Brookhaven National Laboratory, Future Resources Associates, Inc. and the Advisory Committee on Reactor Safeguards (ACRS). The NRC contractors endorsed the quality of the assessment, their comments are provided in Attachment 2.

The Systematic Evaluation Program (SEP) was the most significant confirmation of the Big Rock Point design basis accuracy. The resolution of the Systematic Evaluation Program Topics resulted in procedural and equipment changes. A review of the SEP committed procedural changes was conducted for the development of this response to support the conclusion that the plant configuration was being maintained consistent with the design basis. This self-assessment surveyed the status of the twenty Category 2 IPSAR "Technical Specification changes and procedural development" safety improvements identified in the summary section of NUREG-0828. A review of each item was performed and verified that the commitment was still in place.

In summary, the SEP is considered the design basis validation program for Big Rock Point. A review of the SEP identified procedural changes was performed for development of this response to support the rationale that the design basis requirements are being maintained in the plant procedures. The review confirmed that all twenty of the IPSAR Category 2 items requiring changes to the Technical Specification and procedures were correctly dispositioned. Accordingly, based on the results of the self-assessment and the NRC contractor reviews, the SEP at Big Rock Point successfully evaluated the plant design basis and the enhancements identified in SEP were fully incorporated into plant maintenance, operating and testing procedures.

Station Blackout Rule (SBO)

SBO is an example of an industry initiative that did not affect the plant design basis or result in a major plant modification. Moreover, the SBO initiative is an example of how the results of a comprehensive review of existing plant procedures and plant design basis have enhanced plant operating procedures. The following provides details of this assessment.

On July 21, 1988, the Nuclear Regulatory Commission (NRC) amended its regulations, adding a new section, 50.63, which requires that each light-water-cooled nuclear power plant be able to withstand and recover from a Station Blackout of specified duration. The issue of station blackout involved the likelihood and duration of the loss of offsite power, the redundancy and reliability of onsite emergency ac power systems and the potential for severe accident sequences after a loss of all ac power. The major design differences between Big Rock Point and other licensees responding to this rule were explored in a meeting with the NRC Technical Staff to discuss details explaining the plant's independence with respect to emergency ac power supplies following a loss of offsite power.

As part of this assessment, a test was performed to determine ambient temperatures in the control room without ventilation. Containment heat-up calculations were submitted to confirm the capability of the passive heat sinks. Battery systems required to mitigate the consequences of an SBO (Station Power and Alternate Shutdown) were evaluated and pronounced acceptable in the event of a room heating loss following an SBO. A Station Operating Procedure (SOP) was enhanced to include a caution to maintain a minimum ambient room temperature for the alternate shutdown batteries. A Supplemental Safety Evaluation (SSE) to the Station Blackout Rule dated October 7, 1992, determined that Big Rock Point's response to the SBO Rule was acceptable, upon verification of the capacity of various batteries during extreme cold weather conditions. The required analytical verification was successfully completed.

In conclusion, the SBO Rule review identified no design bases issues and proposed no modifications to the plant. This assessment reaffirmed Big Rock Point's ability to cope with an SBO for many days. The initiative reaffirmed the adequacy of the plant procedures in reflecting the plant design basis. Moreover, the lessons learned from the comprehensive assessment were used to augment the Station Power System Operating Procedures (SOP), Off-Normal (ONP), Alarm (ALP) and Emergency Operation Procedures (EOPs). This also provides an example of how Big Rock Point has revisited its design basis (in detail) to examine why industry initiatives are sometimes not applicable. Refer to Attachment 2 for additional examples; specifically INPO audits, on how Big Rock Point has had to re-evaluate its design basis with respect to procedural controls in order to address generic industry issues.

Instrument Air Generic Letter (GL) 88-14

GL 88-14 is another example of an industry initiative that did not affect the plant design basis. Moreover, the instrument air initiative is an example of how the results of a comprehensive review of existing plant procedures and plant design basis have enhanced plant operating procedures. The following provides details of this review.

Consumers Power Company letter dated February 20, 1989 responded to GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment." The GL requested Consumers Power Company to review NUREG-1275, Volume 2 and perform certain actions to verify design and operation of the Big Rock Point Instrument Air System.

As a precursor to the GL 88-14 response, in 1984 an operational review of the system assuming loss of air was conducted in preparation for the Refueling Outage during which the system would be removed from service for maintenance and modifications. This review ensured that all systems operated as intended during this mode. The results of this review were added as an attachment to the System Operating Procedure for the "Service and Instrument Air System." The results were verified during that Refueling Outage.

As a result of GL 88-14, an additional review of the instrument air system design, including connected air pneumatic accumulators, was conducted. This review has concluded, by analysis, that the design is in accordance with the system's intended function. Big Rock Point Off Normal Procedure ONP-2.2 "Loss of Instrument Air System," which covers symptoms and actions for dealing with loss or decaying pressure in the instrument air system was modified as a result of this effort. As part of this assessment, drawings were created to support plant configuration and valve alignments. In addition to revising ONP-2.2, a monthly surveillance test, T30-60 "Instrument Air System Surveillance" was created to monitor and test the performance of each air compressor and to monitor the quality of the instrument air being supplied.

The Big Rock Point Instrument Air System is not safety-related. All equipment using air can provide their safety-related functions upon loss of air (i.e., spring to close for air operated valves). Procedure enhancements were made to upgrade this nonsafety-related system to more closely reflect current standards. The changes made to the instrument air operating and surveillance procedures improved system reliability.

In summary, the original design bases were determined to be satisfactorily promulgated into plant procedures and again when warranted these procedures were improved.

Motor-Operated Valve Testing - Generic Letter (GL) 89-10

As was the case for SBO and GL 88-14, the motor-operated valve testing program did not affect the plant design basis. Again, the initiative is an example of how a comprehensive review reaffirmed the adequacy of present plant procedures and how the lessons learned were used to enhance plant procedures, in this instance maintenance procedures. The following provides details of this evaluation.

GL 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance dated June 28, 1989, requested that holders of operating licenses and construction permits for nuclear power plants provide additional assurance of the capability of safety-related systems to perform their intended functions by reviewing MOV design basis, verifying MOV switch settings initially and periodically, testing MOVs under design-bases conditions where practicable, improving evaluations of MOV failures and performing necessary corrective action and trending MOV problems. Several supplements have also been issued by the NRC staff to clarify or modify its recommendations.

On July 26, 1995, Big Rock Point notified the NRC that commitments associated with GL 89-10 were complete. The NRC concurred with Consumers Power Company's letter and closed the issue.

During the Phase II inspection, NRC inspectors reviewed the safety-related MOV population and verified that the MOV program was consistent with GL 89-10 recommendations. The program scope consists of 12 MOVs and was unchanged since the initial GL 89-10 inspection. All valves in the program are 4 inch gate valves with the exception of the main steam isolation valve, which is a 12 inch gate valve. No plans were made to revise the Big Rock Point MOV Program scope. In accordance with the program scope accident sequences assuming degraded voltage as well as the expected thermal-hydraulic boundary conditions were evaluated. In fact, because of the detail available in the Limited Scope Simulator, differential pressure transients were analyzed for various accident sequences by running simulator scenarios.

In a letter dated August 21, 1995, the NRC stated that "...You have appropriately reviewed the design basis for program MOVs and verified initial MOV switch settings through testing and use of best available information. Additionally, you tested MOVs under design-basis conditions where practicable, evaluated MOV failures and corrective actions and addressed industry and vendor issues associated with MOV operation."

Because of the GL 89-10 effort, procedures were created for performing valve operator troubleshooting and preventive maintenance. The valve operator overhaul procedures were also revised. As an example of how the design base information was translated into the maintenance procedures, MGP-2 "Setting Limitorque Actuator Limit and Torque Switches" was revised to include new torque switch settings for the valve operators. These new settings were predicated on the design based diagnostic testing. Changes to these settings required Plant Review Committee approval prior to implementation.

In conclusion, GL 89-10 initiated a review of Motor Operated Valve (MOV) design and maintenance. As a result of this program, no valve operators or valves required replacement. Enhancements to maintenance procedures occurred, as with the rest of the industry and these improvements remain in-place today.

Limited Scope Simulator (LSS)

The requirement to provide a plant specific simulator for operator training demanded an assessment of the plant operating procedures as well as the plant design basis. Although the comprehensive assessment confirmed the adequacy of the present plant procedures incorporating the plant design basis, some changes were made. The following summarizes the results of addressing this initiative.

To meet 10 CFR 55.45 requirements, Consumers Power Company in December 1989 began design and construction of the Big Rock Point Limited Scope Simulator. A formal evaluation of the Big Rock Point LSS Facility occurred and was approved for NRC evaluation of Licensed Operators in 1992. In the design and construction of the LSS Facility by Big Rock Point personnel, the plant design bases were revisited.

As an example, a procedure change to Off Normal Procedure ONP-2.20 "Loss of Feedwater/Condensate" resulted from the LSS development. During validation of the control logic for the condensate pumps it was discovered that given a pump trip due to a low hotwell condition, the breaker control circuitry would begin to cycle the breaker open and closed. To avoid the potential of breaker failure due to the repetitive cycling the ONP was revised to have the operator place the breaker control switch to trip to reset the control logic.

The control logic validation task described above, included a detailed review of the plant electrical prints with the system procedures in creating the LSS control logic software. Because these tasks were performed by plant personnel, a circumstantial re-assessment of the plant design basis with respect to procedure adequacy occurred.

In conclusion, the creation of the LSS facility by plant personnel provided another opportunity to review the promulgation of the plant design basis into operating procedures. This effort provided additional evidence that key design basis operational parameters had been correctly incorporated into existing plant operating procedures. Furthermore, this project demonstrated that when procedural inadequacies were identified they were corrected.

Response to Unresolved Item 92-03-01; Emergency Core Cooling System (ECCS) Design Margin

An example of how NRC inspections have required a revisitation of the design basis incorporation into plant surveillance procedures was the safety inspection conducted between January 27 through April 29, 1992 of the engineering and technical support programs by the NRC's Region III staff.

During the course of the inspection, the inspector expressed concerns with the methods used to verify the Emergency Core Cooling System (ECCS) design requirements. Specifically the concerns were with the computer code used to predict ECCS flow rates and ECCS design margin. Inspection report (50-155/92003) issued on May 8, 1992 presented the results of the inspection. Resolution of unresolved items was monitored as unresolved item 50-155/92003-01.

Consumers Power Company responded to this open item on April 15, 1993. This letter provided a brief history of the Big Rock Point ECCS design evolution and an estimate of the ECCS design margin. As part of this assessment, the Florida experimental test data (i.e., ECCS spray test data in a steam environment), the initial ECCS hydraulic analyses developed to support the tests and the original design basis calculations were reviewed.

The inspector also expressed concerns with respect to surveillance testing conducted on the ECCS pumps (at Big Rock Point the initial ECCS water is supplied by the Fire Protection System pumps). Flow testing at the time consisted of verifying each pumps capacity of 1000 gpm at 110 psig, which was the design operating point.

In resolution to the inspector's concern, the pump surveillance test was revised to include a broader range of operating conditions which included the predicted flow conditions from the ECCS hydraulic analysis. On September 30, 1993, Technical Specification amendment number 114 was issued. This amendment added the fire pump head curves used in the ECCS hydraulic evaluations as the acceptance criteria for pump performance. The UFHSR was updated to incorporate the results of these analyses.

In summary, the ECCS design bases were reassessed as a result of NRC inspection 92-03. The reassessment determined that surveillance testing verified the fire pump head curves at the Technical Specification (TS) and UFHSR design operating point. To enhance the monitoring of the fire pumps, the surveillance test and TS's were revised to include a broader range of operating conditions which included the predicted flow conditions from the ECCS hydraulic analysis. This review coupled with the LER-95-001 evaluation (discussed in response (c)) demonstrates how past design basis reevaluations were promulgated into procedures.

Electrical Distribution Safety System Functional Inspection (EDSFI)

Another example of an NRC inspection requiring a revisitation of the plant design that resulted in confirmation of design basis and procedure enhancements was the EDSFI.

This inspection was conducted at Big Rock Point on August 17 through September 22, 1992. With regard to design/licensing basis, the report stated that the team verified conformance with General Design Criteria 17 and 18 and the applicable 10 CFR 50, Appendix B criteria as they applied to Big Rock Point. The team examined the SEP Topics that applied to the Electrical Distribution System.

The review also included the plant Technical Specifications, the Updated Final Hazard Summary Report and appropriate Safety Evaluation Reports to verify that Technical Specification requirements and license commitments were met.

One violation and multiple deviations and unresolved items were identified by the inspection:

> 10 CFR 50, Appendix B, Criterion XI, requires a test program be established to assure that all testing required to demonstrate that components will perform satisfactorily in service is documented and evaluated. Contrary to the requirement, from October 25, 1990 through September 2, 1992, the licensee failed to conduct post modification testing of facility change FC-670 to demonstrate that relays "T" and "CR5", in the Emergency Diesel Generator (EDG) control logic, performed satisfactorily.
>
> Resolution of the inspection items included; documented analyses, procedure changes, new procedures and plant enhancements. These evaluations were incorporated into Chapter 8 of the UFHSR. Engineering procedures 3.1.1.1 Facility Changes and 3.1.1.2 Specification Changes, were revised to ensure that post modification testing would be performed to validate design base considerations.

In summary, the EDSFI, a major NRC inspection program (conducted some 30 years after the commencement of initial power operation), demonstrated the original design capability of the electrical system. Of the dozen open items resulting from the inspection, nearly all involved procedure changes, drawing revisions, information gathering and supporting engineering analyses. Although not required, two hardware enhancements were implemented. These efforts provide assurance that the electrical distribution system procedures are maintaining the plant design basis.

Corrective Action Program (CAP)

The Big Rock Point CAP establishes processes in accordance with Part 50, Appendix B, regarding the identification, evaluation and correction of conditions adverse to quality. Condition report (CR) evaluations have resulted in re-assessments of the Big Rock Point design basis. The two condition reports below, the first of which references the submitted LER, describe how Big Rock Point continues to re-appraise its design basis and capture the significant results in plant procedures.

Basket Strainer (LER-95-001)

During a special test to determine basket strainer differential pressure, the results indicated a plugged strainer. The tested basket strainer provides a final filter of fire protection water as it enters the core spray system. The strainer had been disassembled and inspected several months prior to the testing during a scheduled refueling outage. Subsequent evaluation determined that the basket strainer internals had been installed incorrectly and that inadequate post maintenance testing resulted in the failure to identify the faulted condition. As a result of this event, the 1977 Emergency Core Cooling System (ECCS) design bases were re-assessed. Hydraulic parameters such as pipe segment lengths, geometry-change loss coefficients, elevations, equipment specific loss coefficients, nodal definitions etc. were again re-evaluated. Following the reassessment of the ECCS hydraulic input data, the adequacy of FLOWNET to perform ECCS hydraulic analysis was evaluated using the Engineered Software Inc. FLO-SERIES computer code. FLO-SERIES was found to predict greater flows than FLOWNET for the most limiting ECCS break cases thereby validating the original design basis as well as demonstrating FLOWNET's inherent conservatism's.

In a letter dated May 24, 1995, the NRC issued a Civil Penalty for rendering both core spray systems inoperable. By letter forwarded on July 13, 1995, the NRC reviewed the corrective actions and had no further questions.

As a result of the basket strainer event, the Emergency Core Cooling System (ECCS) design bases were reconfirmed. The assessment reconstituted the design basis for ECCS and promulgated the results into plant procedures. The event occurred as a result of deficient processes; therefore, to preclude re-occurrence administrative, operating and maintenance procedures were revised.

Reactor Vessel Cooldown Rate (C-BRP-96-540)

On May 15, 1996 a self-assessment of the reactor pressure/steam flow chart for a November 1995 reactor scram noted that the Technical Specification 100°F/hr vessel cooldown rate had been exceeded. Several corrective actions were identified as a result of this condition report. Noteworthy tasks included a structural analysis review of the vessel belt line region; a review of past scrams and attendant cooldown rates; and, an energy balance assessment assuming Beginning-of-Core (BOC) conditions.

The belt line structural evaluation included a review of the Combustion Engineering (CE) "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components", published in 1958. The "tentative" design basis evolved from the Naval programs at Westinghouse Bettis and GE Knoll's laboratories and provides the basis for the 1965 ASME B&PV Code Section III. The Big Rock Point normal and off-normal operating conditions were again evaluated against the Section III criteria. Plant scrams from 1970 to the present were reviewed to ascertain the number of cooldown transients. From the number of transients and the GE cooldown criteria, the vessel usage factor was estimated. The investigation concluded that Big Rock Point had not exceeded its design basis. Furthermore the inquiry again demonstrates that Consumers Power Company continually evaluates the plant design basis against the original design criteria.

Off Normal Procedure 13, "Reactor Scrams" and O-TGS-1, "Master Checklist" were revised to provide additional guidance/methods for the operators to ensure the technical specification cooldown rate immediately following a reactor scram is not exceeded.

In summary, the original vessel cyclic design criteria and cooldown rates were reaffirmed as a result of this condition report. Although the design bases were not exceeded for this event, plant procedures were enhanced to reflect the appropriate design basis data.

Response (B) Self-Assessment

As part of the request for information regarding response (b), Big Rock Point has performed an evaluation of currently implemented procedural controls to assess whether design basis requirements in operating, maintenance and testing procedures have been properly maintained or, whether those processes could allow, in the future, inadequate and incorrect translation of these requirements into plant procedures. In addition, three Standard Operating Procedures were reviewed against the UFHSR to provide an appraisal of the degree of agreement between the two documents. Finally, a sample of Technical Specification requirements were selected, compared and correlated to plant procedures. This assessment focused on the following topics:

1. compliance to 10 CFR Part 50 Appendix B,
2. operating procedure adequacy,
3. maintenance processes adequacy,
4. UFHSR to SOP-30 (Control Rod Drive System),
5. UFHSR to SOP-6 (Emergency Condenser System),
6. UFHSR to SOP-5 (Shutdown Cooling System), and
7. Technical Specifications to UFHSR to Procedures.

Compliance to 10 CFR PART 50 Appendix B

The development of the QARM in the late 1980's provided a cross-reference between the Quality Program Description for Operational Nuclear Power Plants, CPC-2A and the plant Administrative Procedures. The QARM is a compendium of the applicable Regulatory Guides, ANSI standards and 10 CFR 50 Appendix B requirements that are committed to by Consumers Power Company in CPC-2A. The use and maintenance of this document provides assurance that the procedure control process remains strong and that old and new requirements are translated, reviewed, approved and controlled by plant procedures. Administrative procedure reviews have identified one recently developed procedure (AP 4.3.3, Planning and Scheduling) that was not entered into the QARM. A CR was written to incorporate this procedure into the QARM.

The Big Rock Point Quality List is used in several processes to identify the significance of Systems, Structures and Components (SSCs). The current review process for the Quality listing component section (i.e., Volume 17 Chapter 4) requires minimal review. Because of the effect the component listing has in the selection of other procedure and work control processes this is a programmatic weakness that will be corrected. A CR was written to improve the Quality List Chapter 4 revision process.

In summary, the Quality Assurance Requirements Matrix (QARM) is an extensive cross-reference which assures regulatory requirements are maintained in plant procedures. Further, the QARM provides assurance that procedure regulatory requirement content is appropriately controlled. Minor discrepancies were discovered between the QARM and administrative procedures that will be corrected. The current review and approval requirements for Q-classification revisions of SSC's create the possibility of inappropriate revisions, which could lead to application of a lesser level of applied controls to maintenance, procurement or engineering activities. The revision process for Chapter 4 of the Q-List will be strengthened.

Operating Procedure Adequacy

The administrative procedures for the Big Rock Point Operations Department allow the use of Temporary Operating Instructions (TOIs) and Procedure Deviations (PDs) in a restricted manner. As a result of a CR, (C-BRP-96-527), a corrective action has already been identified to review the use of TOIs, PDs and Temporary Changes for consistency. An initial review of approximately 150 TOIs and PDs has determined that some could have affected safety-related SCCs. The procedure revision process will be required when safety-related SSCs are affected.

In summary, the Plant operating procedures meet the requirements of Regulatory Guide 1.33. However, the process implemented for providing instructions outside of the normal procedure process will be strengthened.

Maintenance Process Adequacy

Big Rock Point maintenance processes provide controls so that the activities are pre-planned and performed in accordance with written procedures, documented instructions or drawings appropriate to the circumstances. The maintenance process requires determination of the significance of the activity and assigns the controls necessary to assure its proper performance which may include procedures, work instructions or other directives. This process was improved as a result of the Fire System Basket Strainer issue (LER 95-001). The highest level in the work planning process (Level 1) assigned to a work order results in the greatest amount of planning and control. An initial review of Level 1 & 2 CR's (an after-the-fact assessment) has not uncovered any significant problems. A CR has been generated to perform a more detailed assessment of work orders to determine if adequate planning and documentation exists for safety-related maintenance activities.

In summary, the maintenance procedure process was found to comply with Regulatory Guide 1.33.

UFHSR to Control Rod Drive (CRD) SOP-30

The CRD requirements imposed by the plant Technical Specifications and the UFHSR were surveyed for applicable compliance. The results of the CRD investigation showed that an adequate correlation exists between the requirements imposed by the plant Technical Specifications, UFHSR and plant procedure SOP-30; however, weaknesses were discovered and as an example two findings are listed below:

1. Prior to removal of a control blade the specific requirement to remove all four adjacent fuel assemblies needs to be procedurally addressed. A procedure enhancement will be made to implement the requirement that has been addressed with institutional knowledge.

2. The UFHSR will be updated to reflect the actual number of hafnium control blades currently in use. This is not a design issue but simply a clarification of the UFHSR.

In addition, editorial discrepancies were found between the plant operating procedure for the CRD system and the UFHSR. A CR was created to address these discrepancies. Because of these identified weaknesses, a formal review of the UFHSR against plant procedures will be performed.

UFHSR to Emergency Condenser System (ECS) SOP-6

The ECS requirements imposed by the plant Technical Specifications and UFHSR were compared to the system operating procedure. The results of the ECS investigation showed that an adequate correlation exists between the requirements imposed by the plant Technical Specifications, UFHSR and plant procedure SOP-6; however, a weakness was revealed and is summarized below:

In the review of the Emergency Condenser System a weakness was identified. The Emergency Condenser System Surveillance tests, T90-26, "Emergency Valve Operability test," and TR-34 "Emergency Condenser Outlet Valve tests," include an acceptance criteria for valve timing which is different than the valve timing stated in the UFHSR. The valves currently meet all criteria; however, a revision will be made to the plant test procedures such that they are consistent with the UFHSR.

This is one of the data points that lead to our commitment to review plant procedures against the UFHSR.

UFHSR to Shutdown Cooling System (SCS) SOP-5

The SCS requirements imposed by the Plant Technical Specifications (TS) and UFHSR were reviewed against the system operating procedure and discrepancies were discovered. The system operating procedure allows operation of the SCS in such a manner as to maintain the primary coolant system hot and pressurized for an unspecified length of time while work is performed on the secondary side. The TS and UFHSR reference the SCS operation during extended shutdowns but specify use of the main condenser during short duration shutdowns. The TS and the UFHSR do not specifically restrict use of the system as allowed by the system operating procedure. The UFHSR describes the method for controlling the rate of cooling of the primary coolant system that is different from that specified in the system operating procedure.

These editorial discrepancies were found between the plant operating procedure for the SCS and the UFHSR. A CR was created to address the issues. Operation of the SCS in the current manner has no safety significance as the SCS operational mode is not safety related, except as the reactor coolant pressure boundary. This is another data point that led to our commitment to review plant procedures against the UFHSR.

Technical Specifications (TS)-to-UFHSR-to-Procedures

Thirty-one requirements were selected from the Tech Specs for correlation to plant procedures. Applicable sections of the UFHSR were reviewed for the same or similar requirements. Twenty-seven of these requirements were specifically addressed in forty-four plant procedures (including twenty surveillance procedures). The other four requirements have not been specifically incorporated into plant procedures. However, reasonable assurance due to the nature of the requirements exists that they have been met. Procedural enhancements will be made to incorporate the requirements. Because of these identified weaknesses, a formal review of the UFHSR against plant procedures will be performed.

The TS/UFHSR/procedures self-assessment showed good correlation from the TS to procedures.

RESPONSE (B) CONCLUSIONS

Past plant initiatives described how design basis reevaluations were promulgated into procedures.

Administrative procedures, maintenance procedures, operating procedures, surveillance tests and UFHSR maintenance were sampled to evaluate their effectiveness in maintaining system, structure and component design basis configuration and performance. Although areas for enhancement were identified it was determined that the Big Rock Point processes maintain compliance with the regulatory requirements.

Furthermore as a result of this review, additional corrective actions have been initiated to evaluate the need and method for strengthening administrative control processes dealing with maintenance of existing design requirements in plant procedures. Moreover, any proposed upgrades to plant administrative and operating procedures will continue to be evaluated and implemented as appropriate to assure design requirements are being correctly translated into plant procedures.

A self-assessment undertaken as part of the preparation of this response identified some discrepancies between the plant procedures and the UFHSR. Although these discrepancies did not affect system operability or any fundamental design basis assumptions, a more in-depth evaluation of the UFHSR against plant procedures will be conducted to further ensure procedural conformance with the UFHSR. This effort will be completed, including any changes, for the six (6) most risk significant systems by December 31, 1997 and for the rest of the plant systems by December 31, 1998.

Corrective actions have been identified and are described in response (e).

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

SUMMARY

At Big Rock Point there are many reasons for concluding "the system, structure, and component configuration and performance are consistent with the design bases" of the plant. The combination of activities, controls and initiatives described below provides this assurance.

First, the Big Rock Point Plant has not changed significantly from its original design. Since 1962 there have been few major modifications made to the facility. Even as NRC requirements have evolved, only a few major modifications have occurred such as, the redundant core spray system, the reactor depressurization system, and the alternate shutdown building. The major modifications resulting in changes to the design basis have received extensive company and regulator evaluations. Therefore, there have been few opportunities for plant changes to have placed plant performance outside of the design basis and when changes occurred outside the design basis, careful reviews were performed.

The original start-up performance testing conducted by Bechtel and Consumers Power Company determined that equipment, controls and instrumentation were operating satisfactorily and in accordance with plant operating requirements. The acceptance testing did not seek to fully validate all system design basis information, as discussed in Attachment 2, Big Rock Point's design is robust and has been found to provide significant margins in operational performance within the design basis.

The Systematic Evaluation Program (SEP) benchmarked the plant's design against the then current standards and included a review of all major modifications that had been performed at the plant. Design reconstitution/verification for Big Rock Point was addressed by the SEP and Updated Final Hazards Summary Report (UFHSR) efforts. In the context of these efforts independent evaluations and numerous walkdowns have benchmarked the plant design to the then current standards. Big Rock Point has performed no major modifications requiring prior NRC approval since 1990.

Plant modification and configuration changes result in repeated reviews of the design basis and additional analyses are performed when specific information is unavailable. Further activities that have affected the plant design have been reviewed by the Plant Review Committee (PRC).

Finally, a self-assessment determined that the plant configuration with respect to the Integrated Plant Safety Assessment Report (IPSAR) Category 1 commitments was being maintained in accordance with the information presented in the UFHSR. Examples are presented in this response which support the conclusion that Big Rock Point has in the past and continues to assess its design basis with respect to system, structure, and component configuration and performance.

DISCUSSION

This response describes the programs and investigations that have been conducted since the commencement of operation in 1962 which help assure the plant's physical configuration matches the design basis documentation.

In addition for this response, Big Rock Point has performed a self-assessment of the adequacy of the configuration management process described in response (a). The following discussion examines how the system, structure, and component configuration and performance have been maintained consistent with the design basis.

Before entering that discussion it is important to reiterate as discussed in response (a), that the Big Rock Point process to control the configuration of the plant is distributed among various plant procedures. This configuration management process contains practices and procedures that:

1. Control design changes and plant documentation,
2. Promote a culture that emphasizes nuclear safety, self-checking, self-assessments, and performance monitoring of system, structure, and component effectiveness,
3. Help ensure the alignment of the plant with the design basis through administrative and inspection programs,
4. Provide a low threshold corrective action process, and
5. Provide training (from Q-List interpretation to 50.59 writing).

Also response (b) described the programs and investigations that have been conducted since the commencement of operation in 1962 which help assure that the design basis requirements are translated in the present operating, maintenance, and testing procedures, Big Rock Point incorporates those discussions into this response.

Original Big Rock Point Design

The design of the Big Rock Point Plant has not changed significantly since initial start-up. Few major modifications have been made to the facility. These major modifications that have resulted in changes to the design basis have received extensive company and regulatory evaluations.

For example, in December of 1966 the Atomic Energy Commission (AEC) requested that a review of the Emergency Core Cooling System be performed (the original plant design included a single core spray ring sparger). In February of 1970 Consumers Power Company submitted the results of a loss of coolant analysis performed by General Electric and proposed the installation of the redundant core spray line. In June of 1970 the AEC accepted the preliminary design and requested additional information. Following completion of the final design and installation of the approved modification, the Technical Specifications (TS) were revised with the issuance of Amendment Number 26, dated July 27, 1971, officially incorporating the redundant core spray line as part of the plant design.

The original start-up performance tests were performed by Bechtel and Consumers Power Company. These start-up tests were intended to demonstrate that equipment, controls and instrumentation were operating satisfactorily and in accordance with plant operating requirements. The tests were not necessarily intended to verify all design basis information. The documentation of available reports indicate that individual system performance was in accordance with its intended operating requirements.

Further, the original design calculations were quite conservative resulting in a robust plant design. To illustrate, during initial plant design, the operating pressure had not yet been established. During start-up testing the reactor pressure was varied from 800 psia up to 1500 psia. Early in plant life the nominal reactor operating pressure of 1350 psia was selected. The uncertainty in final operating pressure required components be designed to the highest expected operating pressure of 1500 psia. For example, the steam drum safety relief valves were selected to provide approximately 200% relieving capacity when the first valve was set to open at 200 psi above the reactor pressure of 1500 psia.

The emergency condenser is another example of system design margin. The design for the emergency condenser required a 4% (approximately 32×10^6 btu/hr) maximum heat removal capacity. Water storage in the emergency condenser shell was intended to provide four hours of heat removal without makeup. The final design of the emergency condenser included two full capacity tube bundles, which in 1974 were shown to provide a heat removal rate of 41×10^6 BTU/hr/bundle. In re-evaluating the emergency condenser design basis, Big Rock Point acquired the Generalized Containment Model (GCM). The GCM developed by the Department of Energy's Advanced Reactor Severe Accident Program, in cooperation with General Electric has been used extensively to model the GE Simplified Boiling Water Reactor (SBWR) and the Advanced Boiling Water Reactor (ABWR). The code allows the evaluation of local effects analysis such as the impact of non-condensible gases on the emergency condenser tube-to-shell heat transfer film. By employing the GCM, which was successfully benchmarked against the 1974 site-specific test, the capacity of the emergency condenser was shown to increase by over 50% for non-anticipated transient without scram events.

Additional Actions and Initiatives

Several additional actions and initiatives have been undertaken supporting Big Rock Point's conclusions that system, structure, and component configuration and performance are in accordance with the design basis. Examples of those activities and initiatives are listed below and described in detail in the following discussion.

1. Systematic Evaluation Program (SEP)
2. S-Prints
3. Limited Scope Simulator (LSS)
4. Q-List Update
5. Preventive Maintenance (PM) Validation
6. Maintenance Rule
7. Nuclear Performance Assessment Department (NPAD) Assessment
8. Corrective Action

Systematic Evaluation Program (SEP)

The most significant confirmation of the design basis accuracy was the SEP, which is further described in Attachment 2. Due to the detail of the evaluations performed for the SEP, other design bases reconstitution programs were not established for Big Rock Point. The results of SEP were documented in NUREG-0828, which is also referred to as the IPSAR, dated May 1984. Several modifications and procedure changes were generated from the SEP that ensure acceptable system, structure, and component performance in accordance with the plant design basis for those elements of the plant undergoing review. In fact, the SEP involved the review of over 85 regulatory topics. The review of these topics provided assurances of performance in accordance with the plant design basis of the affected system, structures, and components.

To illustrate the depth of the SEP reviews, the evaluation of Topic III-6 "Seismic Design Consideration" was a resource intensive subject. Although not all SEP topics entailed such an effort, the evaluations included fifteen major structures and systems. The anchorage of over fifty equipment items were modified for seismic considerations. To resolve questions concerning the seismic design, a seismic "weak-link" approach was proposed and accepted as a reasonable alternative to providing additional deterministic analyses for topic resolution. The "weak-link" approach identified several outliers that were addressed later through modifications and analyses. The "weak-link" seismic walkdown noted that several motor operated valves were located in close proximity to a structure and system flexibility could permit valve operator damage due to striking the nearby structure. Bumper guards to limit the valve operator movement were installed for several valves to prevent the operators from striking the structure, for two other valves a nearby structure (a work platform) was modified to prevent the valve operators from striking it during a seismic event.

A second example of an SEP driven modification or procedure change occurred in the resolution of topics III-2 "Wind and Tornado Loadings" and III-4.A 'Tornado Missiles". The Alternate Shutdown (ASD) Building was constructed to house equipment required to safely shutdown the plant independent from the control room. Initially the Appendix R modification was a panel located in the post incident heat exchanger room. However to resolve other SEP topics, the panel was changed to a building designed to protect the enclosed equipment from flooding, high winds, tornadoes and tornado missiles. Housed in the building is the 125 vdc battery supply for the MSIV and emergency condenser valves, control switches for valve operation remote from the control room and a portable diesel driven water pump. This pump is capable of providing sufficient makeup to the emergency condenser and was included to resolve questions regarding missile penetration of the screenhouse structure. Procedures were created and revised to take credit for and provide operating instructions for the equipment operated from the Alternate Shutdown Building.

In conclusion, the conduct of the SEP has assured system, structure, and component performance consistent with the plant design basis. A self assessment performed for response (c) concluded that the IPSAR commitments resulting from SEP have been maintained.

S-Prints

In May of 1981 an effort to create more comprehensive plant drawings in addition to the original Piping and Instrument Diagrams (P&ID's) was inaugurated. This initiative provided a tool for assuring that plant configuration was controlled in the context of plant design changes or removal of equipment from service.

New valve-line up drawings (S-Prints) for select systems were created to aid the plant in configuration control. The S-prints added vents, drains and instrument root valves, etc. in greater detail. Development of the S-prints also allowed separating out on single drawings, multiple systems that were condensed onto one drawing in the original P&ID's. The engineering department developed S-prints using existing as-built drawings and P&ID's. The mechanical S-prints were subsequently verified and validated by engineering and operations department walkdowns. The effort completed in November 1983 provided an opportunity to reverify for those systems the Big Rock Point design configuration.

Accordingly, the S-Print project provided an aid in maintaining configuration control thereby supporting continued performance of systems, structures, and components in accordance with the plant design basis. Specifically, these prints are used extensively for determining the scope of control necessary for maintaining proper plant configuration when removing equipment from service.

Limited Scope Simulator (LSS)

To meet the Requirements of 10 CFR 55.45, Consumers Power Company in December 1989 began construction of the Big Rock Point plant LSS, as discussed in response (b). In connection with this effort, a detailed review of the plant design basis was undertaken for primary and secondary systems to the degree necessary to confirm accurate system performance models and logic.

In the design and construction of the LSS by Big Rock Point personnel, the plant design bases were revisited. For example, in the software development phase, the Probabilistic Risk Assessment (PRA) system fault trees were used as the basis to create the pump, valve, system and actuation signal for the FORTRAN control logic subroutines (developed to the contact pair level). This provided a reaffirmation of the plant design and confirmation of the logic models used to evaluate the plant risk during the assessment of Generic Letter 88-20 (Individual Plant Examination for Severe Accident Vulnerabilities).

Another example of design basis use included a previously developed plant specific deterministic model (created to support a recirculation pump modification) that was transformed into the steam secondary side model (e.g., heaters, turbine, hotwell, etc.) for the simulator.

The simulation facility resulted in enhancing the plant design basis knowledge and corroborated that system models reflected critical design basis parameters. This improved understanding of the plant transient response has enhanced the ability of plant personnel to maintain system, structure, and component performance consistent with the original design basis.

Quality (Q)-List Update

The Q-List update is an example of a plant initiative that has provided a usable reference for reviewing design basis requirements. In 1991 Big Rock Point re-assessed the Q-List. The effort focused on the Q-list Chapter 6 system classification in order to clearly identify system safety-related functions, safety-related interfaces, and non safety-related functions. A classification basis for each system was developed with appropriate references to the UFHSR and plant drawings.

This effort improved the ability to identify system safety-related functions, design control interfaces, and non safety-related functions during the work order planning and procurement processes which allow for greater detail and care to be applied to safety related or Q-listed systems, structures, and components. Q-list Chapter 6 is used extensively in making operability determinations. This effort has provided additional support for ensuring the maintenance of the plant design basis.

Preventive Maintenance (PM) Validation

In 1993, an effort was undertaken to review the Big Rock Point preventive maintenance program using a failure modes and effects analysis method for the review. This program was the result of a verbal commitment between senior Big Rock Point management and the NRC, October 27, 1992, and was designated the Preventive Maintenance (PM) Validation Program.

Plant systems were reviewed to identify equipment that had Planned and Periodic Activity Control System (PPACS) established. Data specific to maintenance activities performed on the component were collected. The data included a summary of the PPAC task, maintenance history (3-5 year interval), vendor recommendations, commitments (Technical Specifications, surveillance, UFHSR, and licensing), and a summary of non-PPAC activity maintenance tasks (surveillance tests, maintenance procedures). Subsequently a failure modes and criticality analysis was completed. This included examination of identified failure modes, recommended maintenance from vendors, and commitments against PPAC and non-PPAC activities. In addition, maintenance history was reviewed to determine effectiveness of activity intervals. Either the PPAC was found to be acceptable, or revisions were recommended. Further detail is provided in Attachment 2. Finally, procedures were updated to reflect the results of this effort.

The PM validation program found that plant equipment was being maintained properly. The tools developed for this project improved the capability to monitor the performance of Systems, Structures, and Components (SSC's) in accordance with the plant design basis providing easy to use access to equipment history and related information.

Maintenance Rule (MR)/ Maintenance Rule Workstation(MRW)

The MR initiatives, including the MRW, provided yet further significant mechanisms to maintain control of plant systems, structures, and components in accordance with the design basis. This initiative also included the development of information, data and references that assist in the monitoring of performance of those systems, structures, and components.

To monitor the effectiveness of maintenance at Big Rock Point, the MR was implemented in July of 1996. In order to balance the reliability of SSCs with the availability of SSCs, the extensive PM validation data base (described above) was enhanced, resulting in a PC based Maintenance Rule Workstation (MRW). The MRW includes the MR functional design basis repository of SSC requirements and supporting references that can be used to support performance evaluations and compliance.

The PC based workstation includes:

1. Preventative Maintenance (PM) review,
2. Existing PM's,
3. Operating experience history,
4. Work order's (both current and historical),
5. Modification history (both current and historical),
6. Q-list data,
7. MR documentation (system functions, rankings, performance criteria, MPFFs, Category (a)(1) SSCs, and SSC monitoring), and
8. Corrective action documents (related to systems and /or equipment).

In addition to the MRW, an At-Power Risk model was created to provide a means to review the risk significance of proposed plant configurations. Train definitions and key surveillance tests were provided. Both of these PC based tools incorporate the same design and licensing basis information and as a result aid in assuring plant design basis conformance.

With respect to performance monitoring, in accordance with the MR, Big Rock Point system engineers assimilate design basis information and perform system, structure, and component walkdowns. This assessment is included in maintaining the normal system notebook for selected important systems. Furthermore, periodic "System Health Reports" that describe the system's performance against predetermined criteria, reliability/availability data, maintenance work, temporary modifications, operator work-arounds, and improvement plans are prepared. These reports are submitted to plant management for review. Cumulative trends and industry performance ranking comparisons are subsequently reported to and acted upon by plant management.

With the extensive MR effort, the ability to monitor the performance of systems, structures, and components with the plant design basis has improved. The development of tools for the MR has provided engineering, planners and operators with readily accessible information that is used to ensure configuration and performance are consistent with the design basis.

Nuclear Performance Assessment Department (NPAD)

Audits and other assessment have generally confirmed the adequacy of plant design control processes and products. Issues identified were treated appropriately and generally resolved in a timely manner. Review of NPAD audits, surveillance's and activity monitoring for 1994 -1996 shows that NPAD has been active in examination of both engineering processes and products. The audit program meets the CPC-2A requirements for frequency, subjects and scope of coverage, and has been of sufficient depth to find and report design bases concerns.

Corrective Action Program

As discussed in response (d) the Big Rock Point Corrective Action Program embodies the standard industry practices regarding the identification, evaluation, and correction of conditions adverse to quality. As a result of certain identified conditions, Big Rock Point has taken the opportunity to conduct expanded reviews to reconfirm significant elements of the plant design basis.

Basket Strainer (LER-95-001)

During a special test to determine basket strainer differential pressure, the results indicated a plugged strainer. Subsequent evaluations determined that the basket strainer internals had been installed incorrectly and that inadequate post maintenance testing resulted in the failure to identify the faulted condition. Corrective measures were, of course, taken to address the installation and testing issues. However, in addition, the full corrective action included a comprehensive review of the ECCS performance in accordance with the plant design basis.

As a result of the Big Rock Point condition report, an Emergency Core Cooling System (ECCS) internal white paper was prepared. The white paper revisited the history of the ECCS design, the hydraulic analyses chronicles, and the inherent ECCS design margin. In addition to the written text, the ECCS piping mimics were re-evaluated. These mimics were checked against the July 2, 1979 Industry Experience Bulletin (IEB) (79-14) walkdown data, the Bechtel prints, and the piping isometrics for dimensional accuracy. Subsequent to the historical and layout configuration reconstitution efforts, the original 1977 ECCS hydraulic model was re-evaluated. Hydraulic parameters such as pipe segment lengths, geometry-change loss coefficients, elevations, equipment specific loss coefficients, nodal definitions etc. were again re-assessed.

Following the reassessment of the ECCS hydraulic data, the adequacy of FLOWNET to perform ECCS hydraulic analysis was evaluated using the Engineered Software Inc. FLO-SERIES computer code. FLO-SERIES was found to predict greater flows than FLOWNET for the most limiting ECCS break cases. However, FLOWNET is still used to assess ECCS performance when evaluating TSD-01 results (i.e., fire pump performance data). This ensures additional conservatism in evaluating fire pump acceptance criteria.

The evaluation of this CR resulted in an extensive review of the ECCS design basis developed over 20 years ago. The analyses performed in support of the CR confirmed the original calculations. Furthermore, the updated results reaffirmed the use of FLOWNET in the original analysis. Therefore continued use of the FLOWNET code to monitor Fire Pump Operating Characteristics (TSD-01) data ensures that fire system performance is consistently (and conservatively) monitored against the original plant design basis. No changes to plant configuration were required and performance of the system was in accordance with design bases.

CONTEMPT-LT/28 (C-BRP-96-935)

The evaluation of this CR resulted in an extensive review of the steam line break analyses performed over 15 years ago. The analyses performed confirmed the validity of the original calculations. No changes to plant configuration were required and performance of the containment and electrical equipment qualified components were analyzed and found to be within their design bases.

In Consumers Powers Company's continuing efforts to provide in-house capabilities for accident analyses, a version of the CONTEMPT-LT/28 computer code designed for use on a PC, was installed and a verification/validation effort was initiated. Analyses of past calculations revealed inconsistent results. It was these inconsistencies that led to the decision on October 23, 1996 to voluntarily shutdown Big Rock Point to further investigate the new results.

To summarize tasks performed in support of this CR included: a review of the original mass/energy blowdown and other design base loads; analysis of past and updated input sets with CONTEMPT-LT/28; an examination of the Standard Review Plan and NUREG-0588 with respect to thermal lag analysis criteria; and, an assessment of critical component temperature distributions.

The reevaluation of past steam line break calculations using the revised CONTEMPT computer code showed that Big Rock Point post accident containment temperature calculations were valid and that appropriate temperatures had been used in the environmental qualification of electrical equipment.

Response (c) Self-Assessment

As part of the request for information regarding item (c), Big Rock Point has performed an evaluation of the adequacy of the configuration management program process described in response (a). This evaluation included reviewing the Category 1 items from the IPSAR. The self-assessment reviewed other IPSAR items in support of development of response (b).

Specifically, the SEP was the most significant confirmation of the Big Rock Point design basis accuracy, a review of the SEP identified changes was conducted to provide additional assurance that the plant configuration is being maintained and therefore plant performance is consistent with the design basis. As previously discussed, the resolution of the SEP Topics resulted in various procedural and equipment changes. This self-assessment surveyed the eighteen Category 1 safety improvements identified in the summary section of NUREG-0828. The Category 1 IPSAR items identified equipment changes required by the NRC.

The results of this evaluation indicated that each of the Category 1 equipment changes had been implemented as intended and that those changes had not themselves been altered by subsequent activities. This result indicates that, based on this sample of plant changes, continued control of plant design to assure plant performance is in accordance with the design basis as reflected in the UFHSR.

RESPONSE (C) CONCLUSIONS

The equipment and design of Big Rock Point has not changed significantly. A number of factors have been considered in assessing whether there is confidence that System, Structure and Components (SSC's) at Big Rock Point continue to perform in accordance with the plant design basis. Taken together these factors support the conclusion that plant (SSC) performance continues to be in accordance with the design basis.

Also, as a result of the SEP reviews in the late 70's and early 80's, in concert with the Three Mile Island Action Plan, the Big Rock Point Level III PRA results and the Unresolved Safety Issues program there is substantial evidence that the plant does in fact satisfy its design basis, as supplemented during the SEP. This evaluation benchmarked the plant's design against the then current standards and included a review of all significant modifications that had been performed at the plant. Of note, Big Rock Point has performed no major modifications requiring prior NRC approval since 1990. Further, several initiatives have been undertaken which lend additional support to the conclusion that Big Rock Point continues to assess its design basis with respect to SSC configuration and performance. Finally, it was determined that the plant configuration with respect to IPSAR Category 1 commitments is being maintained in accordance with the information presented in the Updated Plant Final Hazards Summary Report. This review provided further evidence that the plant design, in this case specifically with respect to a set of plant changes, continues to be maintained in a manner that assures performance in accordance with the design basis.

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

SUMMARY

The identification, evaluation, and correction of problems at Big Rock Point is controlled by the Corrective Action Program (CAP). This program is used for resolution of site issues including equipment failures, human performance problems, and design deficiencies. This program has a low threshold for problem identification which encourages a self-assessment culture and utilizes a high level of management involvement throughout the evaluation and correction of problems.

Operability and reportability determinations are an integral part of the process. Conservative reportability and operability decisions were demonstrated by the recent decision to shutdown the plant. The shutdown continued until the results of a containment response evaluation (i.e. design bases confirmation) which indicated problems with Electrical Equipment Qualification could be fully resolved.

Inspection activities conducted by the Nuclear Performance Assessment Department (NPAD), the Institute of Nuclear Power Operations (INPO), the NRC, and insurance underwriters provide additional design review opportunities. Another source of problem identification is through the plant's Nuclear Operating Experience Review program which reviews industry events and information for applicability to Big Rock Point. The CAP incorporates findings from these sources into a single comprehensive system for evaluation and correction.

DISCUSSION

There are many activities and processes at Big Rock Point which provide the opportunity for the identification of problems, deviations, deficiencies and conditions adverse to quality or safety, that may impact plant operation or design. A partial list includes:

1. Identification of issues through ongoing maintenance, operation, work/modification planning, and self-assessment activities

2. Identification of issues through the internal audit program implemented in accordance with 10CFR50, Appendix B

3. Identification through the external audit and evaluation efforts such as Institute of Nuclear Power Operations (INPO) evaluations and assist visits, Nuclear Regulatory Commission (NRC) inspections, and the Nuclear Insurers audits

4. Identification by special review committees whose members are independent from the line organization

5. Identification through the Industry Experience Review program which screens, evaluates and corrects industry identified problems applicable to the plant

6. Identification through semi-annual common cause trend analysis of conditions reported in the CAP

7. Identification through specific reviews included in process controls for activities such as procedure and drawing revisions, preparation of engineering modification packages, revisions to the Technical Specifications and the Updated Final Hazards Summary Report (UFHSR), and through the performance of engineering evaluations

Big Rock Point plant administrative procedures require that adverse conditions, detected in any manner, be documented, evaluated and corrected in accordance with the CAP. Routine equipment problems are typically resolved through the work request/work order process. The work request/work order process addresses equipment operability and immediate actions for equipment control, as well as documenting the work done. A major change was made to the CAP in January 1995 to lower the site's threshold for capturing adverse conditions, to heighten management sensitivity, and to foster a questioning attitude at the site.

In concert with the change to the CAP, Big Rock Point contracted with FPI, International, a recognized industry leader in failure prevention and investigation, to present two courses:

- The first course, entitled "A Comprehensive Course in Advanced Prevention and Reduction of Organizational and Programmatic Failures", was presented to management, supervisors and other selected plant personnel.

- The second course, entitled "A Comprehensive Course in Human Error Reduction", was presented to all Big Rock Point plant employees. The overall course objective was to provide plant employees the skills, in terms of prevention, detection and correction, to reduce human errors in work groups.

The Big Rock Point Administrative Procedure for the CAP establishes the methods for documenting adverse conditions at a low threshold (average of 950 condition reports (CRs) per year which averages out to 5.6 CRs a year per plant employee). The CAP process assures the plant is maintained in a safe condition, determines operability of plant installed equipment according to Generic Letter 91-18 and determines reportability for making timely reports to the NRC in accordance with Technical Specifications, 10 CFR 50.72 and 73, 10 CFR 21, and 10 CFR 73. If necessary, the Shift Supervisor declares equipment inoperable and takes actions required by plant Technical Specifications, including verbal reporting to NRC when necessary. Plant Licensing reviews each CR to assure the correct reportability decision was made, and follows up with any required written reports, including Licensee Event Reports (LER). In addition to reporting required by Technical Specifications and related requirements, there is almost daily contact with the NRC Resident Inspectors, as well as frequent contact with the NRC-NRR Project Manager for Big Rock Point. Resident Inspectors attend many plant status and management meetings, and are informed of significant plant conditions or events at the same time as plant management.

The CAP continues with evaluation to determine the scope, extent, generic implications and cause(s) for the condition, and determination of the appropriate remedial and preventive corrective actions:

- CRs are screened every working day by the Condition Review Group (CRG) which is comprised of representatives from Probabilistic Risk Assessment, Operatic and Licensing. The CRG reviews the CRs to assure that reportability and maintenance ..le applicability determinations are documented and that the CR is assigned an appropriate significance level. CRs which do not represent a significant condition adverse to quality are assigned for evaluation of apparent cause, correction and/or trending. In 1996, the plant completed 298 apparent cause evaluations.

- For significant conditions adverse to quality, the evaluation includes a determination of root cause(s) through various investigative and analytical techniques, identification of implications generic to equipment, systems or processes, and determination of actions necessary to restore the condition to acceptable status and prevent its recurrence. Evaluations of significant conditions are performed or reviewed by dedicated persons specifically trained in root cause analysis techniques. In 1996, the plant completed 78 root cause evaluations.

- A multi disciplinary team evaluation is conducted when an event requires significant resources for recovery and/or there is a significant nuclear safety, environmental, or political impact. The decision to conduct a multi disciplinary team evaluation is made by the Corrective Action Review Board (CARB), the Plant Manager, or the Vice President of Nuclear Operations Department.

Through daily involvement, the Big Rock Point Management Team monitors the CAP, providing the necessary direction and resources to ensure success. Direct involvement includes the CARB (which establishes scope, level of resources and due dates for evaluations, reviews operability an reportability determinations), and the Management Review Board (MRB) (which reviews evaluation results and approves corrective actions, responsibility assignments and concurs with due dates). The MRB meeting is conducted for all significant conditions and/or incidents and also A(1) Maintenance Rule (MR) Category changes. The MRB is, as a minimum, a three-member management quorum which provides a multidisciplinary review of the completed evaluations and MR Category changes.

To effectively manage the CAP a computer software program was developed and implemented to track each identified condition from initiation through evaluation to the completion of corrective actions. This program is on the plant's Local Area Network computer system. This allows plant personnel to search the database for historical information when conducting evaluations and for identifying trends. Action completion is reviewed by management prior to the document being closed out to assure that the specified action(s) has been implemented. Performance indicators for the CAP (including the numbers and age of backlog) are published monthly and discussed by the Big Rock Point Management Team at the monthly performance indicator review meeting. A Semiannual CAP Trend Report is generated to identify adverse trends in plant performance and to provide recommendations for improvements.

The Big Rock Point Administrative Procedure for Reporting Requirements, is used by the Big Rock Point plant staff to address reportability. Overall responsibility for meeting regulatory requirements of the plant rests with the Plant Manager. The Plant Manager designates the Shift Supervisor (SS) to determine initial reportability of the CRs. The Plant Safety & Licensing Department may get involved in the initial reportability. If not, the determination made by the SS is subsequently reviewed by licensing personnel to ensure that the right reportability call was made.

An example of a conservative reporting call was:

- October 23, 1996: Inconsistencies confirmed with NUS/CONTEMPT - LT/28 No-Vent Model computer code. The plant shutdown while the inconsistencies were investigated. Report was retracted following the conclusion that containment response remained unaffected.

The NRC - Office of Nuclear Reactor Regulations (NRR) Project Manager for Big Rock Point communicates with the plant department managers and the Licensing Supervisor by either telephone, voice-mail, and just recently e-mail. This communication, on average, happens every other day, more if the situation requires. Events at the plant are communicated to the NRC - NRR Project Manager in a timely manner. The NRC - NRR Project Manager usually visits the site 2 to 3 times a year; and has replaced the Resident Inspectors when they are offsite.

Big Rock Point is a small site, and contact with the Resident Inspectors usually occurs once a day. Events at the plant are communicated to the Resident Inspectors. The policy states that whenever the On-Call Superintendent is called by the plant, the Resident Inspector is notified.

The Resident Inspectors will usually make tours of the facility and relay their observations to the Plant Safety & Licensing Department for communication to the rest of the plant staff. The Resident Inspectors are also briefed on what the NRC - NRR Project Manager and the plant are working on.

Big Rock Point Administrative Procedures establish the requirements and responsibilities of the Nuclear Operating Experience Review (NOER) Program at Big Rock Point. This procedure applies to the screening, review, evaluation, implementation and disposition of NOER information to provide a mechanism for supplying feedback of operating experience to the plant staff.

At Big Rock Point, NOER information consists of the following reports, notices and letters:

1. INPO Significant Operating Experience Reports (SOERs)
2. INPO Significant Event Report (SERs)
3. General Electric Service Information Letters (SILs)
4. NRC Information Notices (INs)
5. INPO Operations and Maintenance Reminders (O & MRs)
6. Vendor Service Advisories and Notices
7. CRs dealing with Palisades OE Section when deemed applicable by the Palisades-NOERC
8. INPO Operating Plant Experiences (OEs)
9. NRC Bulletins
10. NRC Generic Letters (GLs)

Preliminary reviews are conducted by knowledgeable members of the plant staff, including some Plant Review Committee (PRC) members. The purpose of this review is to provide immediate notification to potentially affected plant areas such that if immediate actions are necessary, they can be taken. The reviews for Licensee Event Reports and 10 CFR Part 21 Reports are dispositioned and resolutions tracked as part of the Corrective Action System.

The evaluation of NOER information is performed by plant personnel with experience of the subject under review. The evaluator is responsible for:

- Analyzing the NOER information

- Providing specific recommendations, and/or initiating actions necessary for resolving the item.

The PRC performs the final review of the evaluation and the actions taken in response to applicable SOERs, SERs, SILs and INs. The Operating Experience Review Coordinator presents the results of the evaluations to the PRC members. The PRC makes recommendations to the Plant Manager for approval/disapproval of actions pertaining to the NOER Program.

Examples of design basis issues evaluated through the NOER system:

- IN 94-10, Failure of Motor-Operated Valve Electric Power Train due to Sheared or Dislodged Motor Pinion Gear Key; Plant procedures were revised to include requirements to restake motor pinion gear key upon reassembly of the motor operated valve.

- IN 95-15, Inadequate Logic Testing of Safety-Related Circuits; Plant procedures were revised to ensure that no part of the logic is overlooked when functionally testing safety-related logic circuits.

- IN 95-37, Inadequate Offsite Power System Voltages During Design Basis Events. The evaluation concluded that the Big Rock Point Electrical Systems are adequate to address the concerns identified in IN-95-37, that could result in inadequate offsite power system voltage during design basis events.

RESPONSE (D) CONCLUSIONS

The CAP is a fundamental process for identification of problems and implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence, and reporting to the NRC. This process has been upgraded to lower the threshold for problem identification and is used for issues regardless of source. The process provides a comprehensive and structured method for analysis and correction of plant problems. Senior plant management involvement is an integral part of the process and has been recognized by INPO as a strength. The system is a readily accessible information source for plant personnel. UFHSR deficiencies have been identified and corrected with this system.

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

SUMMARY

From the information and data presented in this letter and attachments, it is the conclusion of Consumers Power Company that Big Rock Point has effective processes and procedures for maintaining plant design and configuration consistent with the design bases. However, discrepancies have been identified between the Updated Final Hazards Summary Report (UFHSR) and plant procedures. Big Rock Point plans include additional actions to strengthen plant processes and design bases information. The plans are detailed later in this response (e).

The Big Rock Point design bases have evolved since beginning commercial operation in 1962 with the promulgation of regulations. As discussed in Attachment [illegible], the robust design of Big Rock Point resulted from application of the design standards in effect prior to the current regulations and guidance documents. The Systematic Evaluation Program (SEP) reviewed the plant in the late 70's and early 80's in concert with the Three Mile Island (TMI) Action Plan, Big Rock Point Level III PRA results and the Unresolved Safety Issues (USIs) program. This evaluation benchmarked the plant's design against the then current standards and included all modifications that had been performed at the plant. Big Rock Point has performed no major modifications requiring prior NRC approval since 1990. Subsequent evaluations of the plant have shown that the design bases have been maintained.

DISCUSSION

Response (a) describes the processes for maintaining the plant design bases and licensing bases. The attributes of the engineering design and configuration control processes described in response (a) have been effective in meeting NRC requirements and industry expectations. To support engineering design and configuration control processes, administrative procedures were developed in the mid 1970's to assure that processes are in place to translate design bases requirements into working level procedures. The Quality Assurance Requirements Matrix (QARM) was later developed and is used to assure that regulatory commitments made in the Quality Assurance Program Description for Operational Nuclear Power Plants (CPC-2A) are translated into administrative procedures and subsequently into operating, maintenance, and testing procedures.

Response (b) describes the efforts and projects which occurred to align design bases with actual plant operation. Administrative procedures, maintenance procedures, operating procedures, surveillance tests, and UFHSR maintenance were sampled to evaluate their effectiveness in maintaining system, structure, and component design bases configuration and performance.

Consumers Power Company concludes that the plant procedures have adequately incorporated design bases requirements. The SEP project included a review of plant design relative to the General Design Criteria and procedures, as necessary to reflect the Big Rock Point design bases. To provide assurance of continued conformance to the SEP recommendations, Consumers Power Company recently conducted a verification of current implementation of the SEP recommendations impacting plant procedures and found those recommendations to have been appropriately maintained in the procedures.

Numerous additional initiatives subsequent to the SEP have involved further reviews of a broad spectrum of plant/system design bases, including reviews of plant systems found to represent the highest contribution to risk. The Big Rock Point simulator development also entailed an extensive review of design basis parameters and resulted in procedural improvements. These initiatives have generally found that the plant design bases were adequate, and that procedural changes could be instituted to enhance plant operation.

To provide further assurance that plant procedures had adequately captured the design bases, a self-assessment was conducted of procedural conformance to the design bases in the preparation of this response. This review identified no significant problems or examples of non-conformance with the plant design bases. However, the review did uncover questions related to procedural conformance with elements of the UFHSR. Accordingly, Consumers Power Company concluded that additional reviews were warranted to verify the UFHSR provisions were incorporated into procedures.

The above activities have provided assurance that the plant procedures adequately reflect the plant design bases and that the additional actions planned will provide yet further assurance of procedural conformance to the plant design basis.

Response (c) discussed that in addition to the comprehensive SEP, Big Rock Point completed the UFHSR rewrite which was submitted to the NRC in December 1989. Since the submittal, the UFHSR has been maintained by Big Rock Point in accordance with 10 CFR 50.71 and reflects NRC Safety Evaluation Reports (SERs) issued subsequent to the Integrated Plant Safety Assessment Report (IPSAR). To ensure design bases compliance, Big Rock Point utilizes comprehensive processes to assure that plant configuration and performance remain consistent with the design bases. Design reconstitution/verification for Big Rock Point were addressed by the SEP and UFHSR efforts. From independent evaluations to numerous walkdowns, a reassessment of the design bases occurred. Activities affecting plant design bases are reviewed by the Plant Review Committee (PRC) which is composed of senior plant management with significant plant experience. Design bases are thoroughly reviewed and if information is lacking, analyses are performed. No major modifications requiring prior NRC approval have occurred since 1990. The self-assessment summarized in response (c) reviewed the procedural controls for these processes and determined that they maintain the design bases.

Big Rock Point management recognizes the importance of efficient retrievability of licensing and design basis information in ensuring effective processes. Efficient retrievability was first addressed in 1990 with the introduction of a full text searchable version of the UFHSR. This system has since developed into a LAN-based file accessible to all plant members. The files consist of UFHSR, Technical Specifications, and all plant procedures. The Maintenance Rule Workstation (MRW) also provides selected plant personnel with access to significant historical data bases as described in response (c). These tools have become valuable when retrieving design bases data.

Quality training is recognized as a critical attribute to efficiently using and maintaining the design bases and licensing bases information. Plant Administrative Training Procedures contain descriptions of Big Rock Point training programs for Engineering Support Personnel (ESP) training and other design bases related training.

Response (d) concludes that the Big Rock Point Corrective Action Program (CAP) has been successful at identifying and correcting process, performance, and design bases weaknesses. As a result of reengineering the CAP in January of 1995, the plant has seen an increase in condition reports (CRs) related to issues with regards to design basis clarity and compliance. The reasons for revising the CAP were to lower the threshold for capturing adverse conditions, heighten managements sensitivity to issues, and foster a questioning attitude at the site. Prior to changing the program the plant averaged 140 CRs each year for all types of deficiencies; since the reengineering, the plant has been averaging 950 CRs a year. Examples where the new system has been properly used to identify and correct design bases issues include:

- In May, 1995, a CR was written to address a inaccurate description in the UFHSR relating to the Liquid Poison System. The UFHSR stated that "leakage to the liquid poison system is monitored by a low range pressure gauge installed in the main control room." In reality, there are two gauges in the control room that monitor the liquid poison system tank pressure. To correct this situation the UFHSR was revised.

- In August, 1996, a CR was written to address a discrepancy between the UFHSR and Plant Procedures relating to the spent fuel pool. The UFHSR directs that "an investigation of the spent fuel pool configuration shall be required whenever radiation in the sock tank area exceeds 50 mrem/hr." The Plant procedure SOP-44 only requires action when fuel is to be stored in the southern-most three rows of the fuel rack adjacent to the south wall. Plant procedures are being revised to align with the UFHSR.

Review of NPAD audits, surveillances and activity monitoring for 1994 - 1996 shows that NPAD has been active in examination of both engineering processes and products. The audit program meets Quality Program Description requirements for frequency, subjects and scope of coverage, and has been of sufficient depth to find and report design basis concerns. Audits and other assessments generally confirmed adequacy of plant design control processes and products. Issues identified were treated appropriately and generally resolved in a timely manner.

Early in 1994, Consumers Power Company formed a Management and Safety Review Committee (MSRC) to review performance at both its nuclear plants and meet periodically to advise Senior Corporate Management. The MSRC consists of both internal managers and external, highly experienced industry leaders, currently including a retired nuclear utility vice president and a retired NRC regional administrator. The MSRC critically examines Big Rock Points' responses to performance and industry issues, and provides a check on management's prioritization and approach. This critical feedback has served, and continues to, promote appropriately conservative and questioning resolution of important performance and safety issues.

The essential elements of an effective program; comprehensive procedures and processes, a trained and competent staff, problem identification and corrective action, self assessments and independent reviews and audits are in place at Big Rock Point. The comprehensive processes will continue to be verified by the Quality Assurance Requirement Matrix (QARM). The QARM is a computerized report listing the quality assurance requirement bases documents Big Rock Point is committed to; along with identifying the corresponding plant administrative procedure. The processes are workable, which is insured by having working level users assigned as sponsors to initiate and approve changes to the procedures. The deficiency recognition and reporting process has the correct threshold level for insuring problems are identified, with 1069 entries in 199[illegible] And finally, the most essential attribute is the questioning attitude of an involved management team. These attributes provide the basis for the confidence that Big Rock Point has healthy processes and also is actively pursuing resolution of the current weaknesses.

RESPONSE (E) CONCLUSIONS

As summarized above, it is Consumers Power Company's conclusion that our processes and programs are effective in maintaining the configuration of Big Rock Point consistent with its design basis. The corrective action system has been and continues to be used to document discrepancies between the UFHSR and plant processes and programs. Our past reviews and the self assessment for this letter indicate agreement between design bases and plant operation. However, the current assessment indicates that some improvements in processes, correction of discrepancies between plant documents and the UFHSR and additional assessments are required.

COMMITMENTS

The following actions will further assure that plant configuration and operation of Big Rock Point continues to be maintained consistent with the plant design bases and UFHSR.

Process Enhancements

- The QARM is used by Big Rock Point personnel with the understanding that required regulatory commitments are reflected in plant administrative procedures. Some minor clarifications will be made in administrative procedures as well as incorporation of one additional administrative procedure into the QARM.

- The Q-List is used in several processes to identify the significance of SSCs and what procedural requirements apply. Chapter 4 of the Q-List is a component listing that is currently able to be revised with a minimal review process. All other portions of the Q-List require full PRC review with attendant 50.59 reviews as needed. The revision process for Chapter 4 of the Q-List will be revised to strengthen the review process.

- A process which allows the use of Temporary Operating Instructions (TOIs) and Procedure Deviations (PDs) for Operating procedures has allowed these processes to be used on safety-related equipment. Big Rock Point will require the use of a 50.59 review process to determine if safety-related SSCs are affected for all TOIs and PDs. The procedure revision process will be required when safety-related SSCs are affected.

- The maintenance work planning process which allows the use of work instructions and other documents appropriate to the circumstances has allowed these processes to be used on safety-related equipment. The maintenance planning process requires all safety-related SSCs to be planned under the highest/most detailed level of planning unless a PRC approved procedure is used. A more detailed review of completed work orders will be performed to determine if additional controls are required.

Specific Enhancements to Plant Procedures and UFHSR

- In the review of the Emergency Condenser System a weakness was identified. The Emergency Condenser System Surveillance tests, T90-26, "Emergency Valve Operability test," and TR-34 "Emergency Condenser Outlet Valve tests," include an acceptance criteria for valve timing which is different than the valve timing stated in the UFHSR. The valves currently meet all criteria; however, a revision will be made to the plant test procedures such that they are consistent with the UFHSR.

- In review of the Shutdown Cooling System operation a weakness was identified with SOP 5, Reactor Shutdown Cooling System and the description provided in the UFHSR. A revision will be made to reflect the correct operation of the Shutdown Cooling System.

- The results of the CRD investigation showed that an adequate correlation exists between the requirements imposed by the plant Technical Specifications and UFHSR; however, several weaknesses were discovered and the appropriate corrective actions are being taken with regards to revising plant procedures and the UFHSR.

UFHSR Conformance Review

- The self-assessment undertaken as part of the preparation of this response identified some discrepancies between the plant procedures and the UFHSR. Although these discrepancies did not affect system operability or any fundamental design bases assumptions, a more in depth evaluation of the UFHSR against plant procedures will be conducted to further ensure procedural conformance with the UFHSR. This effort will be completed, including any changes, for the six (6) most risk significant systems by December 31, 1997, and for the rest of the plant systems by December 31, 1998.

ATTACHMENT 2
Historical Assessment of Big Rock Point Plant
Design, Modification, Licensing Initiatives and Programs

INTRODUCTION

This attachment provides a historical assessment of the Big Rock Point Plant design, modifications, licensing initiatives, and programs. The discussion includes:

1. A brief description of the plant
2. A summary of some major plant modifications
3. A brief history of major plant licensing issues
4. 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
5. Rationale for the Integrated SEP effectiveness

Plant Familiarization

Consumers Power Company's Big Rock Point Nuclear Power Plant site is located in Charlevoix County, between the towns of Charlevoix and Petoskey, on the northwestern shore of Michigan's lower peninsula. Big Rock Point consists of a direct cycle, forced circulation boiling-water reactor; a power extraction system, and associated service facilities. Consumers Power Company received the Big Rock Point Plant construction permit in May 1960 and the operating license in August 1962. The Big Rock Point reactor is capable of producing 240 megawatts thermal at a nominal operating pressure of 1350 Pia. The electric generating capacity is 75 megawatts at this thermal output.

The Big Rock Point reactor pressure vessel is 30 feet in overall length and 106 inches in diameter. The reactor pressure vessel has 61 penetrations, the largest being the two 20-inch recirc inlet penetrations. The active fuel length is 70 inches and there are 32 bottom entry control blades.

Steam separation and feedwater addition occur in a separate steam drum rather than in the reactor pressure vessel. Subcooled liquid enters the vessel from two constant flow recirculating water pumps near the bottom of the vessel. Each recirculating pump is capable of pumping 6 million pounds per hour of coolant. As the coolant passes through the core, it is heated to a steam-water mixture. Steam baffles, located approximately 6 feet above the top of the active fuel, force the steam-water mixture out six 14-inch risers. The steam-water mixture travels through the risers to the steam drum located, approximately 30 feet above the top of the reactor pressure vessel. The steam drum is 40 feet long and 78 inches in diameter. Three basic functions of the steam drum are: 1) steam separation from the steam-water mixture occurs with turbo-separators and screens; 2) feedwater addition and mixing; and 3) a net positive suction head supply for the reactor recirculating pumps located 65 feet below the drum. The coolant flows from the steam drum to the recirculating pump suction via four 17-inch downcomers. There are no jet pumps in the Big Rock Point reactor pressure vessel and the downcomers are external to the vessel.

The volume of the primary coolant system is large relative to core thermal power and decay heat. Over 35,000 lbm of water covers the core in the reactor and steam drum following a reactor trip. It requires about 2 hours to deplete inventory to the top of the core even if no decay heat removal systems were to function.

Decay heat removal is accomplished using the main condenser, emergency condenser or the shutdown cooling system.

The main condenser is generally used whenever condenser vacuum can be maintained with the steam jet air ejectors. The emergency condenser is used when the main condenser is unavailable and primary coolant system pressure is not low enough (less than 300 psig) to permit use of the shutdown cooling system.

The emergency condenser, similar to the isolation condenser at newer plants, has two internal sets of condensing tubes, each having the capability of removing 100% of the decay heat generated from the reactor following a SCRAM. Because of the decay heat removal capability of the emergency condenser, no transient has caused a demand on the primary system safety valves in the 35 year operating history of the plant. The emergency condenser is located inside containment and is placed in operation either automatically (on loss of ac power, or when reactor pressure reaches 1435 psig) or manually from the control room or alternate shutdown building. After initiation, the emergency condenser can remove decay heat with no immediate operator action required. Following a reactor trip from full power, and with no makeup supply for shell side cooling available, the emergency condenser can remove sufficient decay heat to prevent steam drum safety valve operation (1535 psig) for a period greater than 4 hours. Normally, cooling water to the shell side of the emergency

condenser is supplied automatically by the demineralized water system. However, during loss of ac power conditions, an ac independent means of makeup is available. Fire system water can be supplied to the shell side of the emergency condenser by the diesel driven fire pump through a dc solenoid valve remotely initiated from the control room or the alternate shutdown building. In addition, a portable diesel pump can be manually aligned to supply shell side cooling. The dc power supply for emergency condenser instrumentation and control is located in the alternate shutdown building. Its capacity is such that the Big Rock Point Plant can cope with a station blackout for more than a week.

Other important systems for accident mitigation include the Emergency Core Cooling System (ECCS), which consists of three inter-related systems that provide core cooling. The first of these systems is the Reactor Depressurization System (RDS).

The RDS serves three functions during plant operation. The primary purpose is to provide a rapid depressurization of the primary system to permit operation of the low pressure core spray system when normal high pressure makeup sources fail to provide adequate flow to maintain core coverage. A secondary purpose (and an RDS 'beyond' design base function) of the system is manual primary system pressure control to preclude safety valve actuation for accidents in which primary system heat sinks are unavailable. As a tertiary purpose (and an RDS 'beyond' design base function), the system can be operated to prevent primary system pressurization in accidents where the safety relief valves fail to open.

The system consists of four blowdown lines connected to the main steam header, each sized to pass 144 lb/sec steam at 1350 psig. The blowdown occurs directly into the steam drum enclosure.

The second component of the ECCS is the core spray system. The function of the core spray system is to provide sufficiently distributed spray flow (qualified by plant specific integrated testing) to the core to prevent local fuel cladding failures. The major components of the system are two parallel lines, each having two in-series motor-operated valves, which open automatically on low reactor water level and pressure to provide core spray flow to the core. One train of core spray valves is ac powered from the emergency bus, the other is dc powered from station batteries. The water is supplied by the fire protection system via the diesel driven or electric fire pumps. The suction source for the fire pumps is Lake Michigan as opposed to reliance on a finite condensate or refueling water storage capacity. Because of this unlimited source of injection, if any problems are encountered in transferring to or maintaining core spray recirculation, continuation of, or a return to, the injection mode of core spray allows continued adequate core cooling.

The third component of the ECCS is the post incident system (PIS). The primary function of the PIS, during the recirculation phase of a LOCA is to prevent excess water addition to the containment that could lead to a structural challenge; it also provides for long-term decay heat removal by circulating the water accumulated in the containment through a heat exchanger and then back to the core. The system consists of five suction strainers, two post-incident core spray pumps, one core spray heat exchanger and two motor-operated valves. Because of the size of containment, the requirement to switch from core spray injection to post incident system recirculation could take many hours to days.

The Big Rock Point reactor containment building is a large dry containment, and is significantly different from the current BWR 3, 4 and 5 designs. The Big Rock Point containment is a spherical steel vessel 130 feet in diameter. The sphere extends 27 feet below grade and 103 feet above grade. The Big Rock Point sphere is designed for 41.7 psia internal pressure with the design basis LOCA pressure rating of 37.7 psia. The sphere free-volume is approximately one million cubic feet.

A number of Big Rock Point design features resemble those being considered for Advanced Light Water Reactors (ALWR). These design features are being considered in the advanced designs for their passive reliability and simplicity. These design features, listed in the following table, provide a measure of Big Rock Point's inherent design margin.

PASSIVE ALWR SYSTEMS	BIG ROCK POINT SYSTEMS
BWR Reactivity Control Motor Driven CRDs (In addition to hydraulic). Diverse RPS (RPS & ARI). Passive automatic SLC (ac independent & nitrogen accumulator driven).	**BWR Reactivity Control** Hydraulic CRDMs. RPS. Manual passive liquid poison system (ac independent, nitrogen accumulator, and siphon driven reactor shutdown capability within 75 seconds of initiation).
BWR Reactor Pressure Control 100% relief capacity. Diverse ADS.	**BWR Reactor Pressure Control** 100% turbine bypass capability. 200% relief capacity. RDS. High primary system design pressure (1700 psig) versus operating (1335 psig).
BWR Reactor Inventory Control **High Pressure** Motor driven feedwater (as opposed to steam driven). CRD pump capacity greater than decay heat Isolation condenser (ac independent, 3 day capacity). Reactor clean-up system. **Low Pressure** - LPCI. Gravity injection.	**BWR Reactor Inventory Control** **High Pressure** Motor driven feedwater (as opposed to steam driven). CRD pump capacity greater then decay heat. Emergency condenser (ac independent, 6.5 hr capacity with ac independent makeup). Low volume flow through the cleanup system is allowed by procedure. **Low Pressure** Core spray system (1 train ac independent). Infinite source of water for core spray injection (Lake Michigan). No dependence of engineered safety features on support systems such as service water.
PWR Containment Pressure Control Steam generators. RHR. Passive containment cooling through steel shell Containment spray.	**PWR Containment Pressure Control** Emergency condenser (dc power supply with capacity to last over a week during SBO). Post incident system. Passive heat removal through steel shell (small core/large dry containment). Containment spray.
PWR Combustible Gas Control Igniters. Containment volume capable of keeping hydrogen concentration <10% from oxidation of 75% of the active fuel cladding.	

Containment Isolation Fail safe or dc powered isolation valves.	**Containment Isolation** Fail safe or dc powered isolation valves.
AC Power Non-safety grade diesel generator provided for investment protection.	**AC Power** Single automatic emergency diesel generator. Manual standby diesel generator. Full load-rejection capability (that can maintain plant operation and prevent a reactor trip even with a complete loss of off-site power; although, not demonstrated).

CRDs - Control Rod Drive
CRDM - Control Rod Drive Mechanisms
RPS - Reactor Protection System
ARI - Alternate Rod Insertion
SLC - Standby Liquid Control
ADS - Automatic Depressurization System
LPCI - Low Pressure Coolant Injection
RHR - Residual Heat Removal
SBO - Station Black Out

The EPRI ALWR design criteria require no containment spray; however, the AP600 includes passive sprays (a tank of water) that provide short term flow. The sprays are included only for the non-mechanistic 10CFR100 requirements. The AP600 includes a tank of water outside of containment that provides cooling 1 to 2 days after an accident in order to maintain pressure below design. Water is designed to run down the shell.

Major Modification and Selected Technical Specification (TS) Update Summary

The Big Rock Point Plant received a provisional operating license on August 30, 1962, and began commercial operation on March 29, 1963. A full-term operating license was issued on May 1, 1964. In May 1964, the licensee increased power from 157 MWt to 240 MWt.

The 1962 Provisional Operating License and associated Technical Specifications contained (and still do contain) a large amount of descriptive information regarding the operation of both safety and non safety-related systems at Big Rock Point. Although the current provisions of 10 CFR 50.59 were in place in 1962, the larger scope of information contained in the Technical Specifications necessitated Commission approval over the majority of proposed changes at Big Rock Point. During the late 1960s and 1970s when significant plant changes were completed to address new safety issues, the majority if not all the changes involved Commission design review and approval. The bases for these changes were contained in submittals to the staff and the corresponding issued Safety Evaluation Reports. FHSR update requirements were not in place at that time, however the Consumers Power Company did comply with the reporting requirements associated with 10 CFR 50.59 and included a description of changes made in submittals to the AEC/NRC. Examples of plant modifications prior to and after the issuance of 10 CFR 50.71(e)(3)(ii) include:

1. The reactor thermal shields were modified in the period of September 1964 through September 1965.
2. Because Big Rock Point was susceptible to a single failure (i.e., loss of feedwater) for certain sized LOCAs, the design of the redundant Nozzle Spray System (NSS) began. The final design consisted of a second core spray nozzle, parallel and redundant to the original system, with water supplied from either of the two existing fire water headers or from the core spray recirculation system. This modification was completed in 1971.
3. Installation of two motor operated valves (controlled from the control room) which allows fire water flow to the condenser hotwell for continuous feedwater/condensate pump operation was installed in 1974.
4. In 1975 several component modifications were made based on the exceedance of Equipment Environmental Qualification limits during postulated LOCA conditions.
5. In 1976 Big Rock Point committed to having a second emergency diesel generator operable within 24 hours after a LOCA.
6. The reactor depressurization system was installed in 1976.
7. By 1977 Big Rock Point redesigned both the Ring Spray System and the Nozzle Spray System to account for the effects of steam and/or increased pressures (in and above the upper core region) on spray distribution during a LOCA.

8. The reactor coolant inlet diffuser and a leak in a control rod drive housing were repaired in 1979.
9. Based on the seismic 'weak-links' approach to SEP Topic III-6 several modifications were performed during the mid '80s. Some examples include: installed bumper pads for the redundant core spray valves, MO-7070 and MO-7071; support installation to one vertical column in the outer electrical penetration room; and, a re-evaluation of specific block walls for seismic resistance.
10. Appendix R modifications.
11. Fuel design changes have not occurred since 1986.

In addition to the plant modifications, the scope of the Big Rock Point Technical Specifications was expanded to provide Operability requirements to a larger scope of plant equipment and to add "Administrative" requirements. Included in these additions were:

- LCO's and testing requirements for core spray and containment spray systems.
- LCO's and testing requirements for both ac and dc power systems.
- Operability and surveillance testing for the reactor depressurization system.

In only a few cases in the past 35 years of operation have modifications had a significant impact on the original plant design basis. These modifications include the addition of the backup core spray line and associated enclosure spray changes, installation of RDS, and Appendix R related facility changes. With the exception of Appendix R changes, these modifications were completed prior to the SEP. The Appendix R design was being performed during the conduct of the SEP and the risk assessment. However, insights from each supported the final design and construction of the Alternate Shutdown Building.

Major Licensing and Program History Summary

In the 1960s, the U.S. Atomic Energy Commission's scope of review of proposed power reactor designs was evolving and less defined than it is today. The original Big Rock Point Final Hazards Summary Report (FHSR) was issued on November 14, 1961. The report consisted of two volumes. Volume 1 provided a technical description of the plant consisting of approximately 200 pages. Volume 2 consisted of approximately 50 drawings and 4 topical reports (100 pages). Two revisions to the FHSR were issued dated March 12, 1962 and March 19, 1962. During this period General Electric and Bechtel design documents and calculations were of limited scope and in most cases only supplied to support Staff licensing reviews. These documents were neither controlled nor have they been updated. When possible, Big Rock Point has retrieved original design bases information and has utilized this information when performing plant modifications.

The licensing requirements associated with 10 CFR 50.71(e) and Appendix B to 10 CFR Part 50 were established after Big Rock Point received its Operating License. The provisions of 10 CFR 50.59 were in place during the early years of Big Rock Point operation. Although plant procedures were not required to control the implementation of 50.59 assessments, a review of past correspondence indicates compliance with the license requirements. Further details on this subject are presented in a separate section following this discussion.

The requirements for acceptability evolved as new facilities were reviewed. In 1967, the Commission published for comment and interim use proposed General Design Criteria (GDC) for Nuclear Power Plants that established minimum requirements for the principal design standards. The GDC were formally adopted, though modified, in 1971, and have been used as guidance in reviewing new plant applications since then. Safety guides issued in 1970 became part of the Regulatory Guide Series in 1972. These guides describe methods acceptable to the staff for implementing specific portions of the regulations, including the GDC, and formalize staff techniques for performing a facility review.

In 1972, the Commission distributed "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," now Regulatory Guide 1.70. The Standard Review Plan was published in December 1975 and subsequently updated.

In 1977 the Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) to review the designs of older operating nuclear plants in order to reconfirm and document their safety. The review provided: an assessment of the significance of differences between current technical positions on safety issues and those that existed when a given plant was licensed; a basis for deciding how these differences should be resolved in an integrated plant review; and, a documented evaluation of plant safety. The SEP assessment for Big Rock Point was evaluated in concert with the Three Mile Island (TMI) NRC Action Plan and the Unresolved Safety Issues (USIs) program.

In 1981 the NRC updated the Standard Review Plan which was issued as NUREG-0800. Also during this period, as part of the TMI action plan resolution, Big Rock Point submitted a Level III Probabilistic Risk Assessment (PRA). Consequently the PRA was used to address SEP, TMI, and USI issues as well. The NRC's assessment of these programs was documented in the Integrated Plant Safety Assessment Report (IPSAR) published by the Staff as NUREG-0828 in 1984. As discussed in the abstract of NUREG-0828, the review included:

- An assessment of how Big Rock Point compares with current licensing safety requirements relating to selected issues.

- A basis for deciding how these differences should be resolved in an integrated plant review.

- A documented evaluation of plant safety when the supplement to the IPSAR has been issued.

Big Rock Point considers the SEP a significant design bases validation and has neither undertaken any formal design basis reviews nor reconstitution programs. The NRC's assessment of these programs was performed by independent contractors such as Brookhaven National Laboratory, Future Resources Associates, Inc., and the Advisory Committee on Reactor Safeguards (ACRS). This review was documented in the Integrated Plant Safety Assessment Report (IPSAR) published by the Staff as NUREG-0828 in 1984. As discussed in the abstract of NUREG-0828, the review included an assessment of how Big Rock Point compared with the then current licensing safety requirements. Following the integrated SEP effort, Consumers Power Company formally committed in 1983 to conduct a comprehensive Integrated Assessment (IA) of all open issues (both regulatory and non-regulatory) for Big Rock Point and to develop a living schedule for resolution of important issues. The IA was initially implemented during the evaluation of the SEP. The purpose of the IA was to rank the issues on a relative basis based on perceived magnitudes of reduction in risk or increase in plant availability attributable to their resolutions. The Staff's assessment of the IA was discussed in the IPSAR with additional detail provided at the end of this Attachment.

For example, R. J. Budnitz from Future Resources Associates, Inc., concluded:

"...I very strongly endorse the methodology of this integrated assessment in which the list of SEP issues is considered together with other pending regulatory actions such as generic issues, TMI Action Plant items, and items suggested by the licensee. In my reviews of earlier SEP integrated assessments, I called for just such a broader integrated assessment, and am highly pleased that for Big Rock Point this has been accomplished...."

"...I believe that the existence of the plant-specific PRA has enhanced the usefulness and quality of the Big Rock Point integrated assessment considerably. It is fortunate the utility sponsored the PRA and completed it prior to the start of the SEP review...."

The IA approach was not only used to review industry initiatives but also plant specific enhancements or significant maintenance projects proposed by the plant. When design changes are required they are controlled by administrative processes, part of which is a Plant Review Committee (PRC) review. The understanding of the design basis and implementation of the IA process, which is a part of the plant license, helps ensure the proper application of the design basis to plant modifications. Because the IA was implemented during SEP, the integrated approach to plant modifications has been maintained, since the SEP design bases re-initialization. Major regulatory initiatives affecting design such as Station Blackout, Generic Letter 89-10 (MOVs), etc., have been analyzed as acceptable by Big Rock Point's present design without physical modifications. This further demonstrates the robustness of Big Rock Point's design and illustrates its ability to minimize design bases changes.

S-Prints

In May of 1981 an effort to create more comprehensive plant drawings in addition to the original Piping and Instrument Diagrams (P&IDs) was inaugurated. New valve-line up drawings (S-Prints) for select systems were created to aid the plant in configuration control. The mechanical S-prints added vents, drains and instrument root valves, etc. in greater detail. Development of the S-prints also allowed separating out on single drawings multiple systems that were packed into one drawing on the original P&ID. The mechanical S-prints were subsequently verified and validated by engineering and operations department walkdowns. The effort completed in November 1983 provided a re-assessment of the Big Rock Point design configuration. These prints are used as an aid in maintaining configuration control.

Operational Readiness Assessment

In 1986 an operational readiness assessment of the Post Incident System (core spray, enclosure spray, and post incident recirculation) and the Fire System were evaluated. This assessment evaluated the ability of these systems to meet the system functional design criteria.

QARM

Subsequent to the operation readiness assessment the next significant plant effort (late '80s) involved the development of a Quality Assurance Requirements Matrix (QARM). This matrix provided a detailed road map between 10 CFR 50 Appendix B and CPC-2A (the compendium of the Federal Regulations ranging from Regulatory Guides to ANSI standards from 10 CFR 50 App B that are applicable to Big Rock Point) and the plant Administrative Procedures.

Station Blackout Rule

On July 21, 1988, the NRC amended its regulation in 10 CFR, Part 50. A new section, 50.63, was added which requires that each light-water-cooled nuclear power plant be able to withstand and recover from a Station Blackout (SBO) of specified duration. The issue of SBO involves the likelihood and duration of the loss of offsite power, the redundancy and reliability of onsite emergency ac power systems, and the potential for severe accident sequences after a loss of all ac power.

By letter dated April 17, 1989, March 28, 1990, and October 9, 1991, Consumers Power Company provided its response to the SBO Rule. The NRC issued their Safety Evaluation related to the SBO rule on March 9, 1992. The NRC Staff had determined that Big Rock Point's proposed method of coping with an SBO did not conform to the SBO rule, and listed several recommendations to bring the plant into conformance with the rule. In response, Consumers Power Company met with the NRC Staff on April 10 and May 27, 1992. A revised response was submitted by a letter dated July 3, 1992.

A Supplemental Safety Evaluation (SSE) to the SBO Rule dated October 7, 1992, determined that Big Rock Point's response to the SBO Rule was acceptable dependent on the verification of the capacity of various batteries during extreme cold weather conditions, and the completion of analyses committed to in the July 3, 1992 letter. These commitments were resolved by 1994.

This assessment reaffirmed Big Rock Point's ability to cope with an SBO. This capability is the result of the SEP and the Appendix R program identified plant modifications.

<u>INPO Assessments</u>

1988 INPO Assessment

A 1988 INPO Evaluation Report Finding concluded that some procedures lacked important technical information, used improper formatting techniques, and that key step locations were misplaced. As a result, a procedure upgrade program began in January 1989 and terminated in January 1991. The program focused on rewriting the Standard Operating Procedures (SOPs), the Off-Normal Procedures (ONPs), surveillance tests, Alarm Procedures (ALPs), special operating procedures, and the maintenance working level procedures. The effort concentrated on:

1. Addressing human factor concerns, primarily; formatting, step sequencing, proper application of notes and cautions, and consistent terminology usage.
2. Ensuring technical correctness by comparison to drawings, vendor information, and the plant Technical Specifications.

1993 INPO Assessment

A 1993 INPO Evaluation Report Finding was concerned that water in the reference leg of the Yarway water level instrument would boil if containment temperature reaches the saturation temperature for the existing reactor pressure. Furthermore, the finding recommended that the Emergency Operating Procedures (EOPs) include a relationship between containment temperature and the saturation temperature evaluated at the reactor pressure similar to other Boiling Water Reactor (BWR) EOPs.

To address this finding, the original wide range reactor level design was reviewed in detail, the Emergency Protection Guidelines (EPG) Rev 4 Appendix C application to Big Rock Point was evaluated, the EOP and ONPs assessed, and the performance of the reactor level instrumentation was re-analyzed during Reactor Pressure Vessel (RPV) blowdown and containment heatup for both LOCA short and long term accident phases.

The outcome of this investigation resulted in the addition of a CAUTION statement to the EOPs (e.g., RC/L). This investigation is an example of how Big Rock Point often has to revisit its design basis (in detail) to explain why typical industry practices are sometimes not applicable at Big Rock Point. Because of Big Rock Point's uniqueness (in this case a Pressurized Water Reactor (PWR) containment), re-visiting the plant's design basis is a frequent occurrence.

Besides these INPO assessments, conformance of the plant operating, maintenance, and testing procedures to the plant design basis has been validated as a byproduct of the earlier frequently performed inspections, audits, generic letter driven programs, rulemaking responses etc. Furthermore, a 1994 quality assurance organization surveillance and the self-assessment performed in support of this letter are additional examples of continued procedure examinations.

Instrument Air

A Consumers Power Company letter dated February 20, 1989 responded to Generic Letter (GL) 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment." The GL requested Consumers Power Company to review NUREG-1275, Volume 2 and perform certain actions to verify design and operation of the Big Rock Point instrument air system. In 1984 an operational review of the system upon loss of air was conducted in preparation for the Refueling Outage during which the system was going to be removed from service for maintenance and modifications. This review was essential to ensure that all systems function as intended during this mode. The results of this review were added as an attachment to the System Operating Procedure for the "Service and Instrument Air System." The results were verified during that Refueling Outage.

As a result of GL 88-14, an additional review of the instrument air system design, including connected air pneumatic accumulators, was conducted. This review had concluded, by analysis, that the design is in accordance with the system's intended function. Big Rock Point ONP-2.2 for "Loss of Instrument Air System," which covers symptoms and actions for dealing with loss or decaying pressure in the instrument air system was modified as a result of this effort. As part of this assessment drawings were created to support the plant configuration and valve alignments.

The performance of equipment determined essential to system operation and air quality (e.g., air compressor, aftercoolers, control valves, air dryer, filters, etc) are routinely monitored by the Operations Department as part of plant rounds.

In addition to revising ONP-2.2, a monthly surveillance test, T30-60 "Instrument Air System Surveillance" was created to monitor and test the performance of each air compressor and to monitor the quality of the instrument air being supplied.

Subsequent completed assessments have not identified any components that cannot perform their intended safety function because of instrument air system deficiencies.

FHSR Update

During the SEP, 10 CFR 50.71(e)(3)(ii) was issued. This required in part, that licensees of nuclear power plants subjected to the SEP, file an updated Final Hazards Summary Report FHSR within 24 months following notification from the NRC that the SEP has been completed. In evaluating the benefit of this action as part of the Integrated Plant Safety Assessment for Big Rock Point (NUREG-0828, section 5.3.25), Consumers Power Company proposed to evaluate a method of indexing existing documents to provide a workable substitute versus expending the resources necessary for a complete FHSR update. In a letter dated December 4, 1984, the NRC following review of the Big Rock Point submittal, supported Consumers Power Company's alternative to an updated FHSR. However, just prior to the FHSR cross index submittal in August of 1986, the NRC identified concerns with the developed cross index system. As a result, the Staff requested that Consumers Power Company complete an FHSR update and that a scheduler exemption to allow time to perform this effort was warranted.

In a letter dated December 3, 1986 in support of the scheduler exemption, Consumers Power Company committed to perform the FHSR rewrite. As described within, the goal of this effort was to provide a document which provides a current snapshot of the Big Rock Point design and can be used by the Plant staff to perform 10 CFR 50.59 evaluations. From the NRC perspective the document could be used by their staff to perform reviews of submittals efficiently. As agreed, the document was to follow the existing FHSR outline and format and would not attempt to address the descriptions and degree of detail contained in the Standard Review Plan for Modern Plant Safety Analysis Reports or as described in 10 CFR 50.34(b). The primary purpose of the Updated FHSR (UFHSR) was to reflect the current Safety Analysis Design Bases of Big Rock Point and was accomplished in the following manner:

1. Performing a review of significant facility modifications to provide a current description of Big Rock Point systems and components.
2. Performing a review of Technical Specifications to reflect the current operating requirements of Big Rock Point systems. Conclusions and assumptions from the staff Safety Evaluations for the Amendments were incorporated where appropriate.
3. Performing a review of the final IPSAR (NUREG 0828) and NRC staff Safety Evaluations to the completed SEP topics. Conclusions reflecting design and operation of Big Rock Point were included in the update where appropriate. Other generic actions (i.e. TMI action plan, GL 83-28, etc) were embodied in the same manner.

4. Appropriate sections of the FHSR were updated to reflect the most current transient and accident analyses results. This was not a re-analysis, but a compilation of previous analyses in support of License Amendments, system modifications and generic evaluations on the docket at that time. Many of the analyses resulted from the SEP.
5. Exemptions received from the staff to NRC rules and regulations were also reviewed. Conclusions and assumptions from these exemptions were reflected in the appropriate sections of the FHSR to establish a current licensing basis.

It was also recognized, as with any effort of this magnitude, that inadvertent errors or omissions could be expected. When identified, these are corrected in subsequent submittals in accordance with 10 CFR 50.71(e)(4).

The scheduler exemption was granted in a letter dated March 2, 1987. The submittal of the UFHSR per the agreement above was dated December 22, 1989 and consisted of three volumes and approximately 1000 pages.

Following issuance of the UFHSR in 1990, Big Rock Point drafted an Administrative Procedure for controlling the revisions to the document as required by 10 CFR 50.71(e). The expectation was that the accuracy of the UFHSR would be maintained at the same level of detail as the initial submittal. The Administrative Procedure requires that updates be submitted annually (since changed to refueling outage) and include the following:

1. Plant Modifications
2. Program or Procedure changes
3. License or Tech Spec Amendments including supporting Analyses
4. Exemptions received and supporting Analyses
5. Generic Evaluations and supporting Analyses

Since the UFHSR was issued in 1989, six (6) revisions have been issued and submitted. Although many of the changes were made to correct inadvertent omissions and typographical errors, several of these revisions included significant additions to further clarify the Big Rock Point licensing/design bases. These resulted from both internal and NRC reviews and included the following:

- In 1990 Consumers Power Company conducted an SSFI for the Electric Distribution System and in 1992 the NRC completed a EDSFI. Chapter 8 (Electric Power) of the UFHSR was revised to include the results of the reviews and calculations from these efforts.

- Several additions to Chapter 6.3 (ECCS/PIS) to further describe the design bases of the ECCS and clarification of Big Rock Point's single failure exemption to 10 CFR 50 Appendix K. The additions resulted from both internal reviews as well as concerns from NRC inspections.

- Several open topics from the IPSAR/SEP were still being completed when the UFHSR was issued. As these issues were resolved the FHSR updates reflected the conclusions. In a letter dated April 23, 1996, the NRC acknowledged that the staff had completed its review and resolution of remaining SEP topics. The staff wrote that because the plant has less than 5 years of operation remaining, a supplement to NUREG-0828 would not be issued.

Since its completion, the UFHSR has been accepted as a reasonably accurate source of design and licensing bases information. In cases where detail has been insufficient in identifying the bases of an issue, the references included in the UFHSR provide adequate traceability to documents or submittals necessary to achieve a further understanding.

Generic Letter 88-20 (Individual Plant Examination)

Subsequent to the EDSFI assessment, Consumers Power Company elected to perform a full scope Level I (internal events analysis) and Level II (containment evaluation) PRA as a basis for the Big Rock Point IPE submittal. Big Rock Point's submittal date was delayed two years due to the development of the Limited Scope Simulator (discussed below). The 1994 report conservatively estimated Level I core damage frequency was found to be comparable to that for other plants and was consistent with the results originally presented in the 1981 PRA. Furthermore, Big Rock Point examined Level II issues to the same detail used in assessing the Advanced Boiling Water Reactor (ABWR) design and concluded that the containment was more robust to severe accident challenges than previously thought.

For example, as part of the containment performance assessment, Big Rock Point acquired the Generalized Containment Model (GCM). The GCM developed by the Department of Energy's Advanced Reactor Severe Accident Program, in cooperation with General Electric has been used extensively to model the GE Simplified Boiling Water Reactor (SBWR), as well as modeling of the Advanced Boiling Water Reactor (ABWR). The code allows the evaluation of local effects analysis such as the impact of non-condensible gases on the emergency condenser tube-to-shell heat transfer film. In re-evaluating the emergency condenser design basis the GCM was used. The model which was successfully benchmarked against a 1974 site-specific test, demonstrated that the capacity of the emergency condenser increased by over 50% for a non-anticipated transient without scram events.

Generic Letter 89-10 (Safety-Related Motor-Operated Valve Testing and Surveillance)

Generic Letter 89-10 requested that Big Rock Point assess the capability of safety-related systems to perform their intended functions by reviewing MOV performance. The assessment reviewed the design bases, MOV switch settings, testing of MOVs under design-basis conditions where practicable, improving the MOV failure analysis process, and trending MOV problems (and necessary corrective action). On July 26, 1995, Big Rock Point notified the NRC that commitments associated with Generic Letter 89-10 (Safety-Related Motor-Operated Valve Testing and Surveillance) were complete. The NRC concurred with Big Rock Point's letter and closed the issue.

Limited Scope Simulator (LSS)

To meet the Requirements of 10 CFR 55.45, Consumers Power Company in December 1989, began construction of the Big Rock Point plant-specific mock-up and Limited Scope Simulator. Software development for the work station was also initiated. A total of 655 active instruments and controls had been incorporated as of April 25, 1991. Phase I of Big Rock Point's Simulation Facility was completed including required testing in September 1991. A formal evaluation of the Big Rock Point Simulation Facility occurred and was approved for NRC evaluation of Licensed Operators in 1992. Continued simulator enhancements have been or are planned for the Big Rock Point Simulation Facility (e.g., full turbine and generator controls and responses, enhanced containment model, and instructor console upgrades).

In the design and construction of the LSS by plant personnel, the design basis was revisited. For example, in the software development phase, the PRA system fault trees were used as the basis to create the pump, valve, system, actuation signal etc., FORTRAN control logic subroutines (developed to the contact pair level). This provided a reaffirmation of the plant design and confirmation of the logic models used to evaluate the plant risk during the assessment of GL 88-20. Another example, included a previously developed plant specific deterministic model (created to support a recirculation pump modification) that was transformed into the steam secondary side model (e.g., heaters, turbine, hotwell, etc.). As a result of these efforts, a design issue was identified which required the manual trip of the condensate pumps to prevent breaker cycling following initial hotwell recycle.

Overall creation of the simulation facility has enhanced the plant design basis knowledge as well as improved the understanding of operating procedures.

Electrical Distribution System Functional Inspection (EDSFI)

An Electrical Distribution System Functional Inspection (EDSFI) was conducted at Big Rock Point on August 17 through September 22, 1992. With regard to design/licensing basis, the report stated that "the team verified conformance with General Design Criteria (GDC) 17 and 18, and the applicable 10 CFR 50, Appendix B criteria as they applied to Big Rock Point. The team examined the Systematic Evaluation Program (SEP) Topics that applied to the Electrical Distribution System. The review also included the plant Technical Specifications, the Updated Final Hazard Summary Report, and appropriate Safety Evaluation Reports to verify that Technical Specification requirements and license commitments were met.

Identified Deviations and Unresolved Items

The identified deviations and unresolved items included:

1. EDG installed ventilation system was determined to be undersized.
2. Analysis supporting that the 46 kv line could maintain adequate voltage levels during a Design Base Accident (DBA) with the 2.4 kv voltage regulator out of service was not available.
3. Analysis performed during the inspection concluded that the required Net Positive Suction Head (NPSH) for the diesel generator cooling water pump could be met at high flow rates providing significant lake level decrease or temperature increase had not occurred. Acceptance criteria for lake level and temperature were missing.
4. Suction pressure acceptance criteria identified in the weekly surveillance procedure were deemed too high to assure proper operation of the cooling water system.
5. Suction screen cleaning was not being performed in a timely manner.
6. Wiring discrepancies were discovered between the electrical design drawings and the as-built configuration of the EDG control panels.
7. Control of conduit and cable tray fill requirements was deficient and licensee walkdowns identified some overfilled conduits that did not have a thermal analysis.
8. The unavailability of design information to ensure that installed fuses were properly selected and sized was identified.

Resolution of these items included; documented analyses, procedure changes, new procedures, and plant modifications. These evaluations were incorporated into Chapter 8 of the UFHSR.

Q-List

In 1991 Big Rock Point re-assessed the Q-List. The effort focused on the Chapter 6 system classification. The intent being to clearly identify system safety-related functions, safety-related interfaces, and non safety-related functions. A basis for each system was developed with appropriate references to the UFHSR and plant drawings.

Preventive Maintenance (PM) Validation Program

In 1993, a program (Preventive Maintenance (PM) Validation Program) was created to review the preventive maintenance program using a risk focused method for the review. This program was the result of a verbal commitment between senior Big Rock Point management and the NRC on October 27, 1992. Periodic component activity control (PPACs) performed at Big Rock Point were examined to identify components and systems which had PPACs established. System-by-system, components within the system which had established PPACs were examined and failure modes and effects were identified. Data specific to the maintenance activities were collected. The data included a summary of PPAC task(s), maintenance history (3 to 5 year interval was typically examined, and varied depending on the established PPAC interval or maintenance history), vendor maintenance recommendations, commitments (Technical Specification surveillance requirements, UFHSR surveillance requirements, and surveillance requirements from licensing correspondence), and a summary of non-PPAC maintenance activities (i.e., T-tests, maintenance or operations procedures, operator rounds, etc.)

Subsequently, a failure modes and criticality analysis was completed. Data collected was examined to determine if the failure causes for the identified failure modes would be detected by the identified tasks (PPAC or non-PPAC). Tasks were examined to determine if they satisfied commitment or vendor recommended tasks. Maintenance history (and in some cases, interviews) were used to determine if the frequency met those required by commitments or vendor recommendations. Outliers were those components in "target" systems, which did not have specific PPAC tasks or activities identified for them. For the OUTLIERS, by system, non-PPAC tasks were identified for the components. For those outliers where tasks were not adequate, or the outlier was considered as important (from a risk perspective), the outlier was then evaluated (i.e., data were collected , failure modes and criticality analysis was done, and reviews were obtained.)

Maintenance Rule (MR)

To monitor the effectiveness of maintenance at Big Rock Point, the MR was implemented in July of 1996. In order to balance the reliability of Structures, Systems, and Components (SSCs) with the availability of SSCs, the extensive PM validation data base (described above) was enhanced resulting in a PC based Maintenance Rule Workstation (MRW). The MRW includes the Maintenance Rule functional design base repository of SSC requirements and supporting references that can be used to support performance evaluations and compliance. The PC based workstation includes:

1. Preventative Maintenance (PM) review
2. Existing PM's
3. Operating experience history
4. Work order's (both current and historical)
5. Modification history (both current and historical)
6. Q-list data
7. MR documentation (system functions, rankings, performance criteria, MPFFs, Category (a)(1) SSCs, and SSC monitoring)
8. Corrective action documents (related to systems and /or equipment)

In addition to the MRW, an At-Power Risk model was created and is used by the Work Control Center to provide a means to review the risk significance of proposed plant configurations. Both train definitions and key surveillance tests are provided. Both these PC based tools incorporate the same design and licensing basis information and as a result aid in assuring plant design basis conformance.

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 requirements associated with 10 CFR 50.71(e) and Appendix B to 10 CFR Part 50 were established after Big Rock Point received its Operating License. The provisions of 10 CFR 50.59 were in place during the early years of Big Rock Point operation. Although plant procedures were not required to control the implementation of 50.59 assessments, a review of past correspondence indicates compliance with the license requirements. On October 9, 1963 the first Report of Changes, Tests or Experiments at the Big Rock Point Nuclear Plant was submitted to the AEC. Each described change, test or experiment described in the report was authorized only after Consumers Power Company determined that it did not involve a change in the TS or an unreviewed safety question. The report provided a summary of each of the changes for Commission review.

In a letter from the AEC (March 1971), the Commission stated that in 1970 changes to Appendix B of the regulations became effective and requested Consumers Power Company to submit a description of the corporate Quality Assurance Program for review. In response, a letter to the AEC dated June 24, 1971 provided a discussion of the Consumers Power Company Quality Assurance activities as they specifically applied to the 18 criteria of Appendix B (a Consumers Power Company letter dated March 8, 1973 responded to questions from AEC).

In 1972 Consumers Power Company also provided a copy of the Quality Assurance Policy Manual along with a brief discussion of the eighteen criteria. In the area of design control, the manual required that measures be established to assure that the applicable specified design requirements, such as; design bases; regulatory requirements (including 10 CFR 50.59); codes; and, standards that served as bases for original plant design, construction, and licensing are maintained. The policy required that all safety-related changes be reviewed by the CPC offsite review board to assure "50.59" compliance.

During the same time period that the Quality Assurance Program was being reviewed, Consumers Power Company was also writing Administrative Procedures for Big Rock Point to implement the new QA programs and proposed TS changes to add "Administrative Requirements". These procedures included sections to implement guidelines for "design control" as well as 10 CFR 50.59 and associated reporting requirements. Shortly after issuance of "50.59" procedural guidance, Consumers Power Company recognized that without a current FHSR the regulations in 10 CFR 50.59; "as described in the Final Safety Analysis Report" may have been inadequate to assure a complete review. To improve the process for "50.59" evaluations Consumers Power Company modified its definition of FHSR and Safety Analysis Report to include all docketed information since issuance (but not including) of the Construction Permit. This guidance remained in the Big Rock Point procedures until the time the Updated FHSR was issued.

Following issuance of the UFHSR in 1989, Consumers Power Company developed two procedures (1.11 Safety Evaluations and 6.2 FHSR Management) for implementation of 10 CFR 50.59 and also for control of the Updated FHSR (10 CFR 50.71e). With the exception of minor changes these procedures are still in place. The improved guidance for 50.59 reviews was developed from the guidance in NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations" and also the NRC Inspection and Enforcement Manual; Part 9800. This procedure now follows the guidance of 50.59 and utilizes the Updated FHSR as the basis for performing the evaluations instead of the original FHSR and docket. However, the procedure does inform users that in the event of inadvertent errors or omissions are identified in the Updated FHSR that the original FHSR and docket submittals may have to be referred to achieve resolution. Another important aspect of this procedure is that it describes the requirements for revising the Updated FHSR and also references the plant procedure for controlling revisions to the FHSR. This procedure also provides reference to the Big Rock Point processes and their associated procedures which provide the mechanisms that implement changes to the plant. The processes addressed are as follows:

1. Procedures Program
2. Plant Modifications
3. Facility Changes
4. Equipment Specification Changes
5. Setpoint Changes
6. Jumper, links, and Bypasses/ Temporary Modifications
7. Special Site Tests

Each of the procedures controlling the above processes contain requirements for performing "50.59" evaluations when the change to the system, structure, component, or procedure involves a change as described in the FHSR.

The procedure for FHSR Management addresses plant requirements for revising the Updated FHSR, including review and approval, and submittal to the NRC in accordance with 10 CFR 50.71(e). The issues/changes required to be included in the Updated FHSR revisions are discussed earlier in this letter.

Independent Rationale for the Integrated SEP Effectiveness

As part of the Staff's NUREG-0828 "Integrated Plant Safety Assessment" effort, independent organizations (e.g., national laboratories, consultants etc.) were contracted to review the tasks. The following provides three different summaries of independent assessments of the conduct of the Big Rock Point integrated SEP.

Advisory Committee on Reactor Safeguards (ACRS)

Letter to N. J. Palladino Chairman USNRC from J. J. Ray, Chairman USNRC Advisory Committee on Reactor Safeguards, dated November 22, 1983 (Subject: ACRS Report on the Expanded Systematic Evaluation Program Integrated Plant Safety Assessment of the Big Rock Point Plant).

"...Of the 137 topics to be addressed in the SEP, 29 were not applicable to the Big Rock Point Plant and 23 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the Three Mile Island (TMI) Action Plan. Of the 85 topics addressed in the NRC Staff's review, 53 were found to meet current NRC criteria or to be acceptable on another defined basis and two were resolved during the review. We have reviewed the assessments and conclusions of the NRC Staff relating to these topics and have found them appropriate..."

"...The 30 remaining topics involved 53 issues relating to areas in which the Big Rock Point Plant did not meet current criteria. These issues were addressed by the Integrated Plant Safety Assessment and various corrective actions were considered or proposed by the Licensee and by the NRC Staff..."

"...Our conclusion regarding the SEP review of the Big Rock Point Plant are as follows:

1. The actions taken thus far by the NRC Staff in its expanded assessment of the Big Rock Point Plant are acceptable.
2. We will expect to review the result of the evaluations that are being made and the proposals and schedules for modifications that will result from them.
3. In evaluating the seismic capability, as noted above, assessment of the seismic capacity of weak links will prove to be complex, and care will be required to accomplish an appropriate degree of conservatism (adequate margins) in the light of uncertainties in such capacities. The ACRS expects to review this aspect in detail as part of its evaluation as to whether an acceptable level of risk exists following the modifications..."

Subsequent to the Staff's and the ACRS's concurrence with respect to the seismic weak-link approach, Consumers Power Company submitted the initial evaluation of the seismic 'weak-links' at Big Rock Point in December of 1982.

The general approach of the methodology was to select the design features of the plant most susceptible to an earthquake and those transients felt most important or likely to occur following a seismic event. The system fault tree logic developed for the 1981 risk assessment was simplified as part of the analysis. Passive components and structures were incorporated to the logic models. These seismic 'weak-links' were subsequently updated in November of 1985 and in August of 1989.

Future Resources Associates, Inc.

Letter to C. Grimes USNRC from R. J. Budnitz, Future Resources Associates, Inc., dated November 3, 1983 (Subject: Integrated Plant Safety Assessment, Systematic Evaluation Program, Big Rock Point Plant).

"...I have concluded that the Big Rock Point Plant complies nearly fully with all of the current NRC safety regulations. For a few issues, the compliance is within the intent of the regulation although not with the specific letter of the regulation..."

"...I very strongly endorse the methodology of this integrated assessment in which the list of SEP issues is considered together with other pending regulatory actions such as generic issues, TMI Action Plant items, and items suggested by the licensee..."

"...I believe that the existence of the plant-specific PRA has enhanced the usefulness and quality of the Big Rock Point integrated assessment considerably. It is fortunate the utility sponsored the PRA and completed it prior to the start of the SEP review...."

"...My general impression from the draft report is that plant management at Big Rock Point is effective and competent. First, this impression emerges from reading the report generally. Second, the management took the initiative of proposing the full integrated assessment that includes the TMI issues, the generic issues, and issues desired by the management. Third, the Big Rock Point management has begun to implement a 'risk management' program, based on the lessons from their PRA, that is out in front of the rest of the industry..."

Brookhaven National Laboratory

Letter to C. Grimes USNRC from J. M. Hendrie, Brookhaven National Laboratory Associated Universities, Inc., dated November 14, 1983 (REF: Integrated Plant Safety Assessment Big Rock Point Plant Systematic Evaluation Program).

"...The Draft Integrated Plant Safety Assessment Report on the Big Rock unit is consistent in treatment with previous reports of the Systematic Evaluation. The scope of the report is more extensive than for previous reviews, due to the inclusion, at the licensee's request, of essentially all other outstanding regulatory issues of any significance as well as come plant improvement matters originated by the li... e..."

"...The inclusion in the SEP review of other regulatory matters currently affecting Big Rock Point, and of significant plant improvement projects, allows a coordinated approach to plant changes. Priorities can be assigned, manpower scheduled, and the various jobs done without needless stops and starts. I have been advocating this sort of extended role for the SEP and am pleased to see it in action, in this case at the request of the licensee..."

"...The licensee's probabilistic risk assessment (PRA) study of Big Rock Point is useful as a gauge for the significance of various topics and as a base for the limited PRA exercises of the Staff's consultants. It provides valuable insights into safety issues that would not be available otherwise..."

Letter to C. Grimes USNRC from Herbert S. Isbin, dated October 27, 1983 (Subject: Big Rock Point Plant Review of Draft Report NUREG-0828 Integrated Plant Safety Assessment Systematic Evaluation Program).

"...The Big Rock Point Plant is the ninth plant to be reviewed in the Systematic Evaluation Program (SEP). Unlike previous reviews, this Draft Report presents not only SEP-identified Topics, but also an evaluation of the licensee's integrated assessments including Unresolved Safety Issues (USI), Three Mile Island (TMI) Action Plan Items, and plant-initiated items..."

...ecognition of the plant's small size, location, and specific features, together with the application of the licensee's and the Staff's PRAs, has reduced the number of items that required backfitting. The evaluations presented appear to be reasonable and prudent..."

"...A new and important feature of the Draft is the inclusion of the licensee's integrated assessment of "all" issues with the view of establishing a "living schedule" for resolution and implementation..."

"...I have been favorably impressed by the initiatives taken by the licensee and by the judgments being made by the Staff..."