GPU NUCLEAR

GPU Nuclear, Inc.
One Upper Pond Road
Parsippany, NJ 07054-1095
Tel 201-316-7000

February 10, 1997
6730-97-2011

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

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Facility License No. DPR-16
Response to NRC Request for Information Pursuant to 10 CFR 50.54(f)
Re: Adequacy and Availability of Design Basis Information

This letter contains the GPU Nuclear, Inc. (GPU Nuclear) response to the subject request forwarded by NRC letter dated October 9, 1996. A response was requested within 120 days of receipt of the NRC letter. GPU Nuclear received the letter on October 11, 1996.

The NRC letter requested information on the programs and processes that are applied to operate and maintain the Oyster Creek Nuclear Generating Station (OCNGS) consistent with its design bases including the processes to reconcile deviations in a timely manner. Based on the information contained in the enclosure to this letter and the attachments thereto, GPU Nuclear has concluded our programs and processes are sufficiently effective to provide reasonable assurance that OCNGS is operated and maintained within its design bases. The GPU Nuclear review has identified a number of opportunities to improve our configuration control processes. Only those activities described in Attachment 6 are considered by GPU Nuclear to be commitments.

If you should have any questions regarding this response, please contact Ron Zak of our Regulatory Affairs Department at (201) 316-7035.

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Very truly yours,

T Gary Broughton

T. Gary Broughton
President
GPU Nuclear, Inc.

I, T. Gary Broughton being duly sworn, state that I am President and Chief Executive Officer of GPU Nuclear, Inc., and that I am duly authorized to execute and file this response on behalf of GPU Nuclear. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other GPU Nuclear employees and/or consultants. Such information has been reviewed in accordance with company practices and I believe it to be reliable.

Julienne J Schoffstall

A Notary Public of NJ

JULIENNE J. SCHOFFSTALL
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires June 24, 1997

Attachments

c: Administrator, NRC Region I
NRC Senior Resident Inspector, Oyster Creek
Oyster Creek NRC Project Manager

Enclosure

Oyster Creek Nuclear Generating Station

GPU Nuclear, Inc. Response

10 CFR 50.54(f) Request for Information

Regarding Adequacy and Availability of Design Bases Information

Table of Contents

I. Introduction

By letter dated October 9, 1996, the NRC requested information on the programs and processes that are applied at Oyster Creek to operate and maintain the plant within its design bases, including the process to reconcile deviations in a timely manner. The purpose of this correspondence is to provide an accurate and complete response to this request.

The Oyster Creek Nuclear Generating Station (OCNGS) was issued a provisional operating license on April 9, 1969. The full term license was granted on July 2, 1991 after completion of NRC's Systematic Evaluation Program (SEP) starting in 1977 and continuing through the 1980's. The SEP assessed OCNGS design adequacy with respect to updated design criteria.

Over the approximately 26 years of licensed facility life, many changes have been made and the processes employed to control them have evolved. Recently, the engineering division has been reorganized to align with the equipment reliability and configuration control processes. This organizational structure provides added focus on configuration management.

The information provided in this response, including the attachments, is intended to describe programs and processes as they currently exist. It is not intended to preclude subsequent changes following normal practices, or to require NRC notifications or approvals for such changes other than those currently required. Similarly, the information is not intended to create any new regulatory commitments, except as identified in Attachment 6.

II. NRC Request for Information and GPU Nuclear Response

The NRC request for information was contained in five items:

(a) Description of engineering design and configuration control processes, including those that implement 10 CFR 50.59, 10 CFR 50.71(e), and Appendix B to 10 CFR Part 50;

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

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

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

(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.

In addition to the five items above, the NRC letter requested that GPU Nuclear indicate if a design review or reconstitution program has been implemented or, if not, a rationale for not doing so.

GPU Nuclear used several teams of personnel, knowledgeable of these items to ensure that this response is complete and accurate. A core response team, composed of members from Oyster Creek, TMI-1 and the corporate office, was formed to identify available information relevant to the items. A large body of information was compiled and the most pertinent information was evaluated by senior engineering and quality assurance personnel. The evaluation directly addressed NRC Request Items (b) and (c) and also evaluated the effectiveness of corrective action. When additional data was deemed necessary to address Item (b), a separate effort was undertaken to perform an in-depth review of two systems. The results of this effort were factored into the data evaluation. To ensure an accurate response, another separate effort was conducted to validate the information provided herein. In addition, senior level GPU Nuclear managers and outside consultants met several times to assess the findings and provide feedback.

In responding to the NRC request, GPU Nuclear acknowledges the NRC definition of design bases in 10 CFR 50.2 and the discussion in footnote 4 of the October 9, 1996 letter. Many of the processes described are not limited to this definition, but address other design inputs, sometimes referred to as the engineering design bases.

The response to the NRC request follows:

A. NRC Request - Item (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.

GPU Nuclear Response - Item (a)

The GPU Nuclear Operational Quality Assurance (OQA) Plan is the controlling document which defines the formal and comprehensive plan implementing 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR 50. The OQA Plan is written to meet the requirements of 10 CFR 50 Appendix B and has been approved by the NRC. Within this plan are sections which identify organizations responsible for the implementation of the OQA Plan, and the functional responsibilities of those organizations. Functional responsibilities include control of documents, design, procurement and material, station activities, radioactive materials, corrective actions and nonconformances, and training. Specific sections of the OQA Plan address design control and safety reviews.

Section 2.10, " Safety Reviews", of the OQA Plan establishes the elements of the GPU Nuclear Safety Review Program, incorporating the requirements of 10 CFR 50.59. Major elements of the safety review program are the technical and independent safety review process, and three levels of oversight (Independent On Site Review Group, Nuclear Safety Assessment Department, and General Office Review Board). The program encompasses design change, configuration control, plant procedure change mechanisms, and other activities that could affect change to the plant Safety Analysis Report (SAR) and/or the plant licensing basis including regulatory commitments. In order to facilitate compliance with the requirements of 10 CFR 50.59 regarding changes described in the SAR, the GPU Nuclear Safety Review Process procedure (1000-ADM-1291.01) defines the scope of the SAR. It consists of the updated Final Safety Analysis Report (FSAR), OQA Plan, Emergency Plan and Fire Hazards Analysis Report. GPU Nuclear docketed correspondence which serves as a basis for an NRC Safety Evaluation Report (SER) to support a change to the license or a Technical Specification Amendment (correspondence which is referenced or discussed in the SER) is considered a part of the SAR until incorporated into the FSAR.

Section 3.0, "Control of Document and Records", establishes requirements for the control of written documents and procedures such as the FSAR. Changes to the FSAR are evaluated for acceptability by utilizing the safety review process. These changes result in a formal update to the FSAR pursuant to 10 CFR 50.71(e). The process to update the FSAR is described in a corporate procedure that defines the requirements for administration, frequency, and conduct of the FSAR update.

Qualified engineering and licensing personnel review and concur with proposed changes so that design bases information is adequately translated into the FSAR.

Section 4.0, "Design Control," of the OQA Plan establishes plant design control and documentation measures, including measures to correctly translate applicable design requirements into specifications, drawings, procedures, instructions and material requirements. These measures incorporate the documentation requirements of 10 CFR 50.59.

GPU Nuclear has established a comprehensive set of engineering standards and implementing procedures which complement one another and are used to carry out the engineering design and configuration control requirements established in the OQA Plan. These implementing documents were developed when GPU Nuclear was formed and have been going through periodic revision and redefinition since that time. Procedures and standards have been added, modified and deleted as necessary to address the evolution of our organization, lessons learned through the corrective action process, and changes in regulatory and industry requirements. The existing engineering standards and procedures systematically address aspects of engineering design activities, plant configuration control, design and licensing basis document control and modification control.

The GPU Nuclear system of procedures establishes the methodology by which requirements are translated into plant configuration control documents, design requirements are preserved during the modification process, and plant operating maintenance, and surveillance procedures are updated to reflect the new configuration. In addition, controls exist to so that design basis documents are updated and training programs revised to reflect configuration changes. Programs, processes, and procedures that drive these activities are described in Attachment 1.

Supporting the hierarchy of configuration management documents is a technical support organization which has also gone through a continual improvement process since its establishment. In the summer of 1996, GPU Nuclear integrated the various engineering functions into a new Engineering Division. The Engineering Division is organized along the lines of the Equipment Reliability and Configuration Control Processes developed by INPO under the industry's "Strategic Plan for Building New Nuclear Power Plants" initiative. The reorganized engineering functions reduces the number of interfaces previously needed to implement 10 CFR 50.59, 50.71(e), and Appendix B requirements. This clearer focus on configuration management accountabilities will increase the efficiency of configuration management processes.

B. NRC Request - Item (b)

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

GPU Nuclear Response - Item (b)

Overall, GPU Nuclear has concluded there is reasonable assurance that design bases requirements are translated into operating, maintenance and testing procedures at Oyster Creek. Some areas for improvement have been identified. This conclusion is based on the following:

(1) GPU Nuclear configuration control activities, including translation of design bases requirements into operating, maintenance, and testing procedures, are conducted by a mature organization using processes evolved from lessons learned over time.

(2) GPU Nuclear has sponsored/conducted assessments such as Safety System Functional Inspections (SSFI), and these have identified weaknesses, but have not identified a program failure in the translation of design bases into procedures.

(3) NRC has conducted vertical slice inspections which have also identified weaknesses, but have not identified a program failure in the translation of design bases into procedures.

(4) In preparation of this response GPU Nuclear formed a Data Analysis Team to review past findings and evaluate compliance with design basis requirements. The team concluded that such requirements are generally translated into operating, maintenance and testing procedures; although some discrepancies have been identified, they did not challenge the safe operation of the plant.

Past reviews and the current Data Analysis Team efforts have also identified opportunities for improvement in the current processes for controlling and translating design basis requirements. Therefore, as discussed in the response to NRC Request - Item (e), GPU Nuclear will take a number of steps to strengthen configuration control processes and to identify and correct deficiencies in how design basis requirements are reflected in operating, maintenance and testing procedures.

The following discussion and the referenced attachments provide additional information on the supporting basis for the above conclusion.

1. Procedure Control Processes

Oyster Creek has a well developed system for preparing new operating, maintenance and testing procedures and for controlling changes to existing procedures. This process is described in Section 5 of Attachment 1. Procedure changes are reviewed in accordance with the GPU Nuclear Safety Review Process. Reviewers can draw upon additional resources for reviews as deemed appropriate. This process has been strengthened over the years by lessons learned from experience, and today, GPU Nuclear considers it a mature and effective process.

In addition to the original design bases information, various activities and programs, both self initiated and NRC initiated, have expanded our knowledge of design bases information which has been reflected in operating, maintenance, and testing procedures. Examples include:

- SSFIs conducted by GPU Nuclear, contractors, and NRC personnel,
- Design Basis Document (SDBD) Program,
- Environmental Qualification Program,
- 10 CFR 50, Appendix R Fire Protection Program,
- Vendor Manual Update Program,
- seismic qualification (SQUG) walkdowns, and
- equipment and as-built walkdowns.

2. GPU Nuclear SSFIs and Self Assessment (Vertical Slice Inspections)

Several vertical slice inspections have been conducted/sponsored by GPU Nuclear. These have concluded that the systems reviewed were capable of performing their design bases functions. SSFIs and a Service Water System Operational Performance Inspection indicated that operating, maintenance and testing procedures were adequate to ensure system performance consistent with the design bases. A summary of five GPU Nuclear initiated inspections is contained in Attachment 3.

3. NRC Vertical Slice Inspections

The NRC has also conducted vertical slice inspections at Oyster Creek. These have also indicated that the systems reviewed meet their design bases. Design control programs were judged adequate. These observations confirmed that processes in place were generally effective in translating the design bases into plant procedures. A summary of the three NRC inspections is contained in Attachment 4.

4. Data Analysis Team Review

A Data Analysis Team was formed to review recent reports and assessment documents, and to use the results to evaluate how well the design basis requirements are incorporated into Oyster Creek operating, maintenance and testing procedures. The Data Analysis Team at Oyster Creek was composed of senior engineering and quality assurance personnel familiar with Oyster Creek. The team focused their attention on the review of reports and assessments from the most recent years. The team reviewed over 130 documents. The list of which can be found in Attachment 2. Approximately 600 observations were itemized, evaluated, and entered into a database for analysis. The principle focus was on items which had some safety significance directly related to NRC Request - Items (b) and (c).

The Data Analysis Team found a limited number of observations in these documents related to procedural compliance with design bases. As a result, a separate effort was initiated to obtain more observations regarding the effectiveness of design bases incorporation into operating, maintenance and testing procedures by conducting an in depth review of two safety systems. The selection of two systems was considered sufficient based upon their representative nature and complexity. The two systems reviewed were Core Spray and Automatic Depressurization System (ADS). The results showed that design basis requirements were generally incorporated correctly into the procedures; some discrepancies were found, but none raised operability concerns and corrective actions are in progress.

Analysis of the data and discussions among Data Analysis Team members resulted in the following conclusions about how well design basis requirements are translated into operating, maintenance and testing procedures at Oyster Creek:

- Documented reviews imply good performance in translating design basis requirements into procedures.

- The many reviews/audits/inspections reviewed as part of this effort (Attachment 2) did not find a programmatic problem in translating design basis requirements into procedures.

- Reviewers looking deeply in specific areas (ESW SWSOPI, ADS SSFI, NRC Inspection Reports, etc.) have generally found good performance.

- The results of the review of FSAR and Technical Specification requirements related to the Core Spray and ADS systems provided an independent confirmation that such requirements are generally translated into appropriate procedures.

- Our procedure change process requires writers and reviewers to consider the SAR (which includes the FSAR which is readily available to them on the computer with full word search capability). Reviewers are requalified biennially.

- The procedure change process is weak on linking non-modification changes which may affect design basis back to updating the design basis documents. Two examples identified temporary modifications which were made permanent and acceptance criteria changes.

- Writers are not required to be trained in procedure writing or the SAR. Trained reviewers are relied upon to catch differences between the procedure being reviewed and the SAR. Experience has shown examples of reviewers not commenting on changes which do not adequately consider/reflect the SAR. This is indicative of a weakness in training in this area.

C. NRC Request - Item (c)

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

GPU Nuclear Response - Item (c)

Overall, GPU Nuclear has concluded there is reasonable assurance that system, structure, and component (SSC) configuration and performance are consistent with the design bases at Oyster Creek. Documented reviews imply GPU Nuclear is generally effective in maintaining plant configuration and performance consistent with design bases, although these reviews also demonstrate that opportunities for improvement exist in the processes used to maintain SSC configuration and performance. This conclusion is based on the following:

(1) GPU Nuclear configuration control activities, including maintaining plant SSC configuration and performance consistent with the design bases, are conducted by a mature organization using processes evolved from lessons learned over time.

(2) GPU Nuclear has sponsored/conducted assessments such as Safety System Functional Inspections (SSFI) and these have identified weaknesses, but have not identified a program failure in the area of maintaining plant configuration and performance consistent with the design bases.

(3) NRC has conducted vertical slice inspections which have also identified weaknesses, but have not identified a program failure in the area of maintaining plant configuration and performance consistent with the design bases.

(4) In preparation of this response, GPU Nuclear formed a Data Analysis Team to review past findings and evaluate compliance with design bases requirements. The team concluded that GPU Nuclear is generally effective in maintaining plant configuration within the design bases; although some discrepancies have been identified, they did not challenge the safe operation of the plant.

Past reviews and the current Data Analysis Team efforts have also identified opportunities for improvement in the current processes for controlling and translating design basis requirements. Therefore, as discussed in the response to NRC Request - Item (e), GPU Nuclear will take a number of steps to strengthen configuration control processes and to identify and correct deficiencies in maintaining plant SSC configuration and performance consistent with the design bases.

The following discussion, and referenced attachments, provide additional information on the supporting basis for the above conclusions.

1.0 Design Control Processes

Oyster Creek has a well developed system for controlling plant design of SSCs. This process is described in Section 5 of Attachment 1. Design changes are reviewed in accordance with the GPU Nuclear Safety Review Process. Reviewers can draw upon additional resources for reviews as deemed appropriate. This process has been strengthened over the years by lessons learned from experience and today GPU Nuclear considers it a mature and effective process. Self assessments and corrective action processes will continue to be used to make further improvements.

In addition to the original design bases information, various activities and programs, both self initiated and NRC initiated, have been conducted to establish and/or verify the configuration of Oyster Creek and to reasonably assure that SSC configuration and performance are consistent with the design bases. Examples include:

- SSFIs conducted by GPU Nuclear, contractors, and NRC personnel,
- Design Basis Document (SDBD) Program,
- Surveillance Testing,
- Inservice Testing Program,
- Inservice Inspection (ISI) Program,
- Plant Trip Review,
- Post Maintenance Testing,
- Startup and Test Activities,
- Maintenance Rule Program,
- Quality Assurance Audits and Assessments,
- Environmental Qualification Program,
- 10 CFR 50, Appendix R Fire Protection Program,
- Vendor Manual Update Program,
- seismic qualification (SQUG) walkdowns, and
- equipment and as-built walkdowns.

2.0 GPU Nuclear SSFIs and Self Assessment (Vertical Slice Inspections)

Several vertical slice inspections have been conducted/sponsored by GPU Nuclear. These have concluded that the systems reviewed were capable of performing their design bases functions. SSFIs and a Service Water System Operational Performance Inspection indicated that SSC configuration and performance was

maintained. A summary of five GPU Nuclear initiated inspections is contained in Attachment 3.

3.0 NRC Vertical Slice Inspections

The NRC has also conducted vertical slice inspections at Oyster Creek. These have also indicated that the systems reviewed meet their design bases. Design control programs were judged adequate. These observations confirmed that processes in place were generally effective in maintaining SSC configuration and performance. A summary of the three NRC inspections is contained in Attachment 4.

4.0 Data Analysis Team Review

A Data Analysis Team was formed to review recent reports and assessment documents, and to use the results to evaluate how well SSC configuration and performance are maintained consistent with the design bases. Details on the Data Analysis Team can be found in the response to NRC Request - Item (b).

Analysis of the data and discussions among Data Analysis Team members resulted in the following conclusions about how well SSC configuration and performance are maintained consistent with the design bases at Oyster Creek.

- The many reviews/audits/inspections reviewed as part of this effort did not find a programmatic problem in the area of maintaining plant configuration and performance consistent with design basis.

- Reviewers looking deeply in specific areas (Core Spray SSFI, ADS SSFI, NRC Inspection Reports, etc.) have generally found good performance.

- Strengths from the SSFIs on Core Spray and ADS indicate that modifications have preserved the design basis of the systems.

- The modification and turnover processes have procedural requirements to update the SAR as the plant is modified. Experience has shown that the requirements are almost always successful in updating the design basis documentation as the plant is changed.

- The safety review process is weak on recognizing when non-modification changes may affect design basis. Two examples identified temporary modifications which were made permanent and acceptance criteria changes.

D. NRC Request - Item (d)

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

GPU Nuclear Response - Item (d)

GPU Nuclear has several processes available which fulfill the functions specified in this request. At Oyster Creek, the Plant Deviation Report process is the most frequently utilized corrective action process for identification of problems and implementation of corrective actions, including root cause determinations. The process, using a graded approach, determines the extent of problems and actions necessary to prevent recurrence. Corrective actions are input and tracked to completion by various action item systems. When the need to report to the NRC is identified, GPU Nuclear utilizes its notification and Licensee Event Report (LER) procedures to meet the reporting requirements of 10 CFR 50.72 and 50.73.

The corrective action program is described in a corporate level procedure that has GPU Nuclear-wide applicability. While there are several corrective action systems currently in place throughout the company, each approved corrective action system has the attributes required to reasonably assure that nonconforming conditions are properly characterized and resolved, and that actions are taken to prevent recurrence when required. The systems in use include requirements for timeliness, trending, escalation to management when required, and verification of corrective actions.

Material nonconformances identified in installed equipment are controlled and tracked using a Material Nonconformance Report (MNCR). Receipt Deficiency Reports (RDR) provide reasonable assurance that nonconforming procured material will not be inadvertently used.

External organizations, such as audit teams from the Cooperative Management Audit Program and evaluation teams from INPO and the National Academy for Nuclear Training provide for organizational learning concerning industry operating experience related to corrective action issues. Oversight of the corrective action program is provided by periodic audit by independent certified auditors. Also, senior industry personnel make up GPU Nuclear's General Office Review Board (GORB) which reviews safety related activities at GPU Nuclear, including corrective actions.

Attachment 5 provides a more detailed description of the corrective action program.

E. NRC Request - Item (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.

GPU Nuclear Response - Item (e)

Based on a significant body of data and information, GPU Nuclear concludes that Oyster Creek programs and processes are sufficiently effective to reasonably assure that the configuration and operation of the plant are maintained within its design bases.

In the course of the reviews leading to this conclusion, some deficiencies have been found and areas warranting improvement have been identified. These are being acted upon. None, however, suggest to us fundamental weakness in the GPU Nuclear programs and processes – on the contrary, the overall finding from the evaluations is that the programs and processes are effective, and generally achieve their intended objectives.

GPU Nuclear's confidence in this conclusion is based on the following:

- The processes and procedures are sound and mature, having been used, refined and tested for many years.

- The GPU Nuclear organization is mature and capable; there is a strong and committed management team and a corporate culture and history of excellent performance. Also, recent organizational changes have clarified and simplified interfaces, particularly in the configuration control area.

- Internal oversight and assessments have generally confirmed the effectiveness of the GPU Nuclear configuration control activities.

- Externally sponsored evaluations, by INPO and NRC, have yielded conclusions consistent with the internal ones.

These have been addressed in the above responses to NRC Requests - Items (a) through (d). Further supporting information is summarized in the following paragraphs:

Processes
The response to NRC Request - Items (a) and (d) describe Oyster Creek processes to maintain the plant configuration and design bases. These processes include overall configuration management, configuration change, safety review and other

processes. Oyster Creek's corrective action systems, periodic self-evaluations, and process improvement reviews monitor performance in this area and provide recommendations for improvement. Overall, these processes have been generally effective in maintaining the design bases of the plant.

Safety Culture

The effectiveness of Oyster Creek's processes and programs is enhanced by the management philosophy communicated to employees on a continuing basis. The GPU Nuclear working environment encourages employees to freely report safety concerns and then provides for prompt evaluation and resolution to those concerns. This expectation has been facilitated by the establishment of the Ombudsman position for safety concern reporting, specific employee training on how to report safety concerns, and establishing a low threshold for deviation reporting. Consistent with this philosophy, the GPU Nuclear employee incentive compensation program places high value on plant safety and excellence (50% for SALP, INPO, and GPU Nuclear - Nuclear Safety and Compliance Committee indicators).

Engineering Division Reorganization

Recently, the GPU Nuclear engineering organization was changed, with the goal of improving processes and performance. The new engineering organization, implemented in the Summer of 1996, is structured around the two processes of configuration control and equipment reliability. This applies increased focus on configuration control activities including design bases issues.

Process Review and Refinement

As a follow-up of the engineering reorganization in the Summer of 1996, reviews have been conducted and are ongoing for major work processes. Some improvements have been implemented and more are expected. Many of these are directly related to the design bases and configuration control areas. These include the following:

- Integration of existing design bases program and reconstitution project efforts into the functional organization under the Configuration Control Department.

- Development of an integrated process procedure for SDBD periodic review and maintenance.

- Assignment of a process owner, within engineering, for programmatic control of SDBDs and the FSAR.

- Development of specific review guidelines for the SDBDs and FSAR, to provide verification that key statements and parameters are reflected in operating, surveillance, and maintenance procedures. This review will be completed on a periodic basis during the FSAR biennial update for plant changes **(Attachment 6, Commitment 1).** This will provide a process to more thoroughly assess the accuracy of FSAR and SDBDs and their translation into procedures. Further, the effort will include documenting which statements and parameters are reviewed and those deficiencies discovered, so that this information may be used to guide subsequent reviews.

- A one-time detailed verification of the Oyster Creek FSAR will commence in 1997, to examine parameters and statements and verify that they are adequately translated from design bases documents into plant procedures **(Attachment 6, Commitment 8).** This review will include the same guidelines developed for the upgraded FSAR biennial review except it will be more comprehensive.

- Conduct personnel training on design bases issues and the new FSAR update process **(Attachment 6, Commitment 5)**.

Design Basis Reconstitution
In 1989 the plant initiated a formal design bases reconstitution program. It is described in Attachment 7, which identifies the systems, structures, and components (SSCs) included in the program as well as the method of prioritization and current status. The program has resulted in the development of 16 design bases documents, procurement of design information from General Electric and Burns and Roe, and an ongoing program to evaluate and gather design bases information as needed.

Activities to Verify Oyster Creek Configuration Management
Over the years multiple activities have been conducted to verify the configuration of Oyster Creek and its adherence to design bases. Some of these are:

- Environmental Qualification (EQ) Program
- Seismic Qualification (SQUG) Program
- Equipment and component configuration walkdowns
- Vendor Manual Program
- Appendix R Fire Protection Program compliance

Activities like these provide additional assurance that the plant configuration is being maintained, and in some cases they establish and verify design bases. For example, the EQ program established and documented the design bases for the

environmental qualification of electrical equipment, verified pertinent aspects of plant configuration, and produced a program to maintain these design requirements.

Internal and External Evaluations
GPU Nuclear and outside organizations have conducted design bases and configuration control reviews/inspections at Oyster Creek. Some of these are discussed in the response to NRC Request - Items (b) and (c), including major reviews such as the NRC Safety System Outage Modification Inspection (SSOMI), Service Water System Operational Performance Inspection (SWSOPI), and NRC Electrical Distribution System Functional Inspection (EDSFI). In general, these reviews have indicated that Oyster Creek's design bases and configuration are well maintained. Where deficiencies have been identified by these assessments, corrective actions have been taken (although in some cases, not as expeditiously as desired).

Data Analysis Team Review
In preparing to respond to the NRC 10 CFR 50.54(f) Letter, GPU Nuclear assembled a Data Analysis Team to examine the assessments conducted over the past decade and develop an overall sense of their conclusions. Recommendations and action items from a sampling of corrective action systems, internal and external audits, design bases reconstitution, and major configuration control and design inspections were reviewed, cataloged, and assessed. The Data Analysis Team concluded with reasonable assurance that GPU Nuclear is generally effective in maintaining Oyster Creek procedures and plant configuration consistent with the Oyster Creek design basis. The team also identified areas for improvement. Attachment 6 summarizes the commitments which GPU Nuclear is making to address deficiencies and upgrade processes.

Corrective Action Effectiveness Review
Based on the Data Analysis Team review of corrective action effectiveness and other relevant information, GPU Nuclear has concluded that we are generally effective in completing corrective action for important items. Some observations in the corrective action area include (i) items entered into a formal corrective action system were closed out more quickly than those entered into some other task tracking system; (ii) changes in corrective action systems since the early 1990s have consolidated and streamlined the corrective action process into fewer systems; and (iii) items not entered into a formal corrective action system are sometimes not resolved in a timely manner. Based on these observations, GPU Nuclear will consolidate resolution tracking for design bases open items to improve timeliness of resolution **(Attachment 6, Commitment 2)**.

NRC Design Inspection at TMI
GPU Nuclear has evaluated the findings from the recently completed NRC Design Inspection at TMI-1 for applicability to Oyster Creek. Improvement will be made in the control and update of calculations and Technical Data Reports (TDR), and in setpoint basis documentation / references **(Attachment 6, Commitments 3 and 6).**

Summary
GPU Nuclear concludes that the programs and processes in place are generally effective in maintaining the Oyster Creek configuration consistent with its design bases. Oyster Creek was designed in the early 1960s and placed in commercial operation in 1969. For that reason, its engineering design bases is not as readily identifiable as for later-vintage nuclear units. Nevertheless, the necessary design basis is available from the aggregate of the FSAR, licensing documentation, System Design Basis Documents, engineering documentation from the architect engineer and NSSS supplier, and other sources.

In summary, based on experience in operating the plant since 1969 and as demonstrated by the results of internal and external assessments, GPU Nuclear has concluded that there is reasonable assurance that the Oyster Creek design bases are understood, available to those who must access them, and generally reflected in plant documentation. Further, this same experience indicates that our processes are sufficiently effective in applying the information to provide reasonable assurance that plant configuration is being maintained consistent with the design bases.

F. NRC Request - Design Review or Reconstitution Program

"...indicate whether you have undertaken any design review or reconstitution programs, and if not, a rationale for not implementing such a program. If design review or reconstitution programs have been completed or are being conducted, provide a description of the review programs, including identification of the systems, structures, and components (SSCs), and plant-level design attributes (e.g., seismic, high energy line break, moderate-energy line break). The description should include how the program ensures the correctness and accessibility of the design bases information for your plant and that the design bases remain current. If the program is being conducted but has not been completed, provide an implementation schedule for SSCs and plant-level design attribute reviews, the expected completion date, and method of SSC prioritization used for the review."

GPU Nuclear Response - Design Review or Reconstitution Program

GPU Nuclear has been conducting a Design Bases Reconstitution Program for the last eight years. For Oyster Creek, the program has resulted in the development of 16 design basis documents (1 partial) including the reconstitution of event based design information, and the performance of five Safety System Functional Inspections (Attachment 7, Enclosure 1). The Design Basis Documents prepared by GPU Nuclear focus on answering the question, "why" functional or parametric requirements for a given system were selected. The program also includes the identification and procurement of Supporting Design Information from both the NSSS supplier (General Electric) and architect engineer (Burns & Roe). We currently have procured Burns & Roe Oyster Creek Specific Design Information and are continuing to selectively procure and catalog Supporting Design Information from General Electric as needed.

The GPU Nuclear Design Bases Reconstitution Program has been effective in improving our knowledge and availability of our most important design basis information. This has resulted in:

1) insights into fundamental design philosophies and intents;

2) identification of gaps in the design documents, design process documents and supporting design information which have been selectively filled; and,

3) identification of changes to operational procedures.

The objectives of our program are being met by locating design bases design information for selected systems, and capturing and disseminating this information using the Design Basis Document. The salient features of the program are:

- The program was initiated as part of an overall upgrade of the GPU Nuclear Configuration Management Program that resulted from a detailed GPU Nuclear self assessment of Technical Support in 1987-1988; a pilot program was conducted in 1989 to test the approach to be used to manage, staff, conduct, document, and review the results of the reconstitution process.

- Some of the critical criteria used to select the systems for inclusion in the Design Bases Reconstitution Program and to formulate the schedule for completing the program were safety significance, risk significance based on probabilistic risk assessment techniques, retirement of key original designers, cost of system maintenance, existence of system challenges, and support for system engineers and system performance teams.

- GPU Nuclear created its own specification for the content and approval of design basis documents. The focus is to determine the bases for why structure, systems and components exist. Teams of GPU Nuclear and NSSS and/or AE personnel have been utilized.

- Accessibility to Design Bases Information has been upgraded through the preparation of Design Basis Documents and improved knowledge of and accessibility to original files.

- The Design Basis Document is developed as a configuration control document in which resides the engineering design bases and associated supporting design information with referencing to source documents and important design documents for the selected system. It may also include the modification history and significant historical aspects of the system.

- The Design Basis Document is reviewed for completeness and accuracy by a review team prior to approval and release for use.

- The Design Basis Document is to be maintained current by utilizing the GPU Nuclear Engineering Change Documentation (ECD) process.

The results achieved to date include:

- Original engineering design bases and associated supporting design information retained by GE Nuclear Energy has been located and are currently being procured on a selective basis. Oyster Creek design and construction documentation is currently retrievable.

- Original engineering design bases and associated supporting design information previously retained by Burns & Roe have been procured, evaluated, partially reorganized and incorporated into our record management system.

- 16 Design Basis Documents (one partial) covering 20 systems and one topic have been issued for use. Five SSFIs / self assessments have been completed.

Attachment 7 provides a summary description of our ongoing Design Bases Reconstitution Program including a list of those systems for which the development of a Design Basis Document is under consideration (Attachment 7, Enclosure 3). GPU Nuclear plans to continue to utilize a process involving multi-discipline reviews and prioritization criteria to continuously assess and define the future scope and output of Design Basis Reconstitution Program activities **(Attachment 6, Commitment 4).** Using this process, GPU Nuclear has identified Design Basis Reconstitution activities for 1997 (Attachment 7, Enclosure 2; **Attachment 6, Commitment 7).**

Attachment 1

Configuration Control Processes

This attachment provides a detailed presentation of GPU Nuclear's Configuration Control Processes and Organization and the relationship of requirements (10 CFR 50 Appendix B, 50.59, 50.71(e)).

Configuration Control Processes

1.0 ENGINEERING ORGANIZATION

In the Summer of 1996, GPU Nuclear integrated their various engineering activities into a new Engineering Division. The new division was organized along the lines of the Equipment Reliability and Configuration Control processes. INPO facilitated the development of these processes under industry's "Strategic Plan for Building New Nuclear Power Plants". The Engineering Division organization is described in outline format below.

1.1 Equipment Reliability Process

This process is consolidated under a single organization at each plant site and is managed by the Director - Equipment Reliability. The organization is made up of five groups that have responsibility for the equipment reliability process. The following is a list of these groups.

- System Engineering
- Components
- Equipment Reliability Programs
- Shift Engineers
- Process Computers

1.2 Configuration Control Process

This process is consolidated under a single organization at each plant site and is managed by the Director - Configuration Control. This organization is made up of five groups with responsibility for the configuration control process.

- Configuration Maintenance
- Electrical Power and Instrumentation
- Mechanical and Structural
- Modifications
- Design and Drafting

1.3 Corporate Resources

In addition to the site based engineering organizations, corporate engineering resources are available to support and augment the plant's technical needs. The

Director - Engineering Support manages this organization. This organization is composed of the following eight groups with the following roles in the organization.

- Components and Programs
- Nuclear Fuels
- License Renewal
- Safety and Risk Analysis
- Decontamination and Decommissioning
- Mechanical and Structural
- Electric Power and Instrumentation
- Projects

1.4 Chemistry and Materials

Also part of the engineering organization within GPU Nuclear is Chemistry and Materials. The Chemistry and Materials Director manages this organization. This organization manages the Chemistry Laboratory and Materials Laboratory in Reading Pennsylvania to support both nuclear plants as well as the non-nuclear installations within GPU. They also provide NDE / ISI services and environmental and chemistry support at both plant sites.

2.0 QUALITY ASSURANCE AND ASSESSMENTS

2.1 Quality Assurance

The Operational Quality Assurance (OQA) Plan has been established to meet the requirements of 10 CFR 50 Appendix B. Within the OQA Plan, Section 2.0 describes the scope and approach of the Quality Assurance Program, QA Program review and control, classification of SSC, safety reviews and responsibilities.

2.2 Design Control

Section 4 of the OQA Plan requires that measures be established and documented to assure that the applicable design requirements which include design bases, regulatory requirements, codes and standards are correctly translated into specifications, drawings, procedures and instructions.

These control measures include the organization structure, design bases research, material selection, communication during design development, deficiency correction, quality verification , applicability of commercial grade items, design

verification, computer code procedures, design change control, procedural control of documents and plant awareness of changes. The Plan also defines the responsibilities of the Directors whose activities are affected by the design control process.

The corporate engineering procedures and standards form the body of implementing documents to carry out the design control measures established in Section 4 of the OQA Plan. Site specific procedures also serve as implementing documents, but refer back to the corporate procedures as the controlling documents.

2.3 Assessments

Section 10 of the OQA Plan establishes that a program of assessment will be conducted by the Nuclear Safety Assessment Department. The assessment program will combine elements of assessment, monitoring and audit to assess the adequacy of performance for activities within the scope of the OQA Plan.

A portion of the audit function within the OQA Plan is to plan and schedule systematic, proceduralized audits of documents which prescribe methods and provide the technical requirement for activities within the scope of the Plan.

A program for monitoring of activities within the scope of the OQA Plan has been established and executed by the GPU Nuclear Safety Assessment group. Monitoring is used to establish adequate confidence levels that activities within the scope of this plan are being performed in accordance with the QA Program requirements and plant administrative controls. Monitoring is performed on a graded approach with the degree of monitoring performed based typically upon the status and safety importance of activities, extent of previous experience, thoroughness of overall coverage, uniqueness of testing or operating activities and trending data.

3.0 CONFIGURATION CONTROL PROCESSES

The configuration control processes which implement the requirements set forth in 10 CFR 50.59, 10 CFR 50.71(e) and Appendix B to 10 CFR part 50 are divided into categories of the OQA Plan. Chapters 3.0 through 9.0 control the functional activities within the plan's scope, outside the areas of organization, Quality Assurance, and Assessment. These chapters are titled:

- Control of documents and records
- Design Control

- Procurement and Material Control
- Control of Station Activities
- Control of Radioactive Wastes or Materials
- Control of Corrective Actions and Nonconformances
- Control of Training

4.0 IMPLEMENTING DOCUMENTS

The requirements of 10 CFR 50.59, 50.71(e) and Appendix B are addressed by GPU Nuclear in a tiered approach. The OQA Plan is at the highest tier, written to comply with the requirements, followed by corporate procedures on the next tier and plant specific procedures as required. Prepared as implementing documents, corporate and site specific procedures are generally grouped into families .

5.0 Configuration Management Details

Details on activities which maintain, change and control plant configuration consistent with licensed plant configuration and the design bases are presented below. This section contains references to individual procedures from the procedure families and to plant specific procedures. While not all inclusive the section provides a summary description of key processes.

5.1 Safety Review Process

Scope:
The requirements of 10 CFR 50.59 are implemented in accordance with the GPU Nuclear Safety Review Program as defined and controlled by the GPU Nuclear Operational Quality Assurance (OQA) Plan. Plant Technical Specifications also specify safety review program requirements. The elements of the safety review process which implement these requirements are specified in procedure 1000-ADM-1291.01, 'GPU Nuclear Safety Review Process'. This procedure is controlled by the OQA Plan. The applicability and scope of the safety review procedure requires appropriate evaluation and review of plant changes or activities that affect the SAR, are within the OQA Plan scope, or are included for other reasons.

Description:
The safety review process as defined in procedure 1000-ADM-1291.01 requires that the documented evaluations and reviews for these changes or activities appropriately consider possible adverse affects on nuclear safety or safe plant operation and the unreviewed safety question criteria specified in 10 CFR 50.59. The safety review process envelopes design change, configuration control and

plant procedure change mechanisms that could affect change to the plant SAR and / or the plant licensing basis including regulatory commitments. The activities that are included in the safety review process are identified in the safety review matrix that is part of the procedure; however, any plant change or activity which affects the SAR requires a safety review. Procedure 1000-ADM-1291.01 contains the safety review matrix for Oyster Creek.

The Safety Determination and 50.59 Review is used to first determine whether the change or activity has any potential adverse impact on nuclear safety or safe plant operations, involves a change in the license, Technical Specifications, or is within the scope of 10 CFR 50.59. If any of the above are applicable, then a safety evaluation is required. The screening document requires Responsible Technical Reviewer (RTR) review and approval.

A safety evaluation must be developed if after completion of the safety determination document, one is determined to be required. The purpose of the safety evaluation is to determine if the activity or change will result in an unreviewed safety question, a technical specification change, or will impact on nuclear safety. By addressing the possible impact on nuclear safety, the safety evaluation is broader than the requirements of 10 CFR 50.59 and it also encompasses activities outside the scope of 10 CFR 50.59, which still may present safety concerns.

The safety evaluation is prepared using the guidance and requirements established in the safety review process (1000-ADM-1291.01) to address and document safety related issues associated with the activity or change. The safety evaluation is also subject to cross disciplinary reviews as appropriate and RTR review. It then must be reviewed by the Independent Safety Reviewer (ISR).

Two additional elements of the safety review process involve a technical review and, as necessary, an independent safety review of changes or activities as appropriate. These reviewers are required to be certified as RTR and ISR and qualified through initial safety review process training with periodic requalification training. Levels of independence from the change or activity being evaluated are specified, and thus provide an additional level of assurance that potential adverse impact on nuclear safety or safe plant operation are identified and that the design and licensing basis is preserved. These training programs incorporate 10 CFR 50.59 examples and industry lessons learned, as well as process improvement areas identified through self assessment activities.

Procedure 1000-ADM-1291.01 is the controlling document for safety reviews within GPU Nuclear, although there are site specific and division specific versions of this procedure that invoke the same requirements and refer to the corporate procedure.

Level of Reviews:
The Responsible Technical Reviewer reviews the safety evaluation for technical, adequacy and applicability from an independent point of view. The Independent Safety Reviewer reviews the safety evaluation for completeness, and accuracy regarding safety significant aspects of the evaluation.

5.2 Quality Classification List

Scope:
To provide accurate readily accessible information on each component's quality requirements.

Description:
The quality classification list is an established database within our maintenance work control database (GMS2), which establishes the quality classification for plant components and sub-components. The quality classification provides a basis necessary for the procurement of spare parts, for installation of new components and for maintenance control. The quality classification process is controlled by procedure 5000-ADM-7313.02 (EP-011), 'Methodology for Preparing the Quality Classification List'. A six digit code is used to establish the following attributes for each component:

- Functional Class: categorizes components based on functional importance to plant operation and safety, such as pressure retaining portions of reactor coolant pressure boundary.
- Seismic Classification: such as operable during and after SSE.
- Functional Mode: categorizes components based on their condition of operation to achieve the required function, such as automatic operation, or maintain integrity.
- QA/QC Basis: specific or special nature of a component, such as component reliability, or complexity in fabrication.
- Safety Classification: Nuclear Safety Related, Regulatory Required or Other.
- QA/QC Requirement: If QA or QC activities are required or not.

Procedure 5000-ADM-7341.03 (EP-035), 'Component Identification Requests / Component Data Base Maintenance and Updating', provides the guidance, requirements and reviews necessary for changing, adding or updating the QCL database.

Level of Reviews:
The QCL checklist, which is the input document for updating the database, is approved by an independent design reviewer.

5.3 Design Basis Documents

Scope:
Design basis documents consist of System Design Basis Documents (SDBD), System Design Descriptions - divisions 1 & 2 (SDD) and Modification Design Descriptions - divisions 1 & 2 (MDD). The method for the preparation, revision, review and approval of these documents is covered in procedure 5000-ADM-7313.01 (EP-005), "Modification and System Design Descriptions and System Design Basis Documents".

Description:
System Design Descriptions are developed for new systems being added to the plant configuration. The Division I documents the design bases for the system and identifies the detailed requirements (including licensing and regulatory requirements) the system must meet along with the technical bases for those requirements. The Division II is the detailed description of the system as finally designed and installed. It includes detailed system description, system performance characteristics, system arrangement, instrumentation and control, system interfaces, limitations, set points, precautions, testing requirements and operations.

Modification Design Descriptions are developed for modifications to systems and contain the same type information as SDDs only they reflect the changes to the systems they affect. MDDs must also reflect the consistency of the design with the established design bases for the systems they affect. MDDs are not required for all modifications, however when the MDD is not used, similar information will be contained in the Configuration Change Document. Once the modification is installed existing SDDs or SDBDs that are affected must be updated to include the new information either by revision or engineering change document posting.

System Design Basis Documents have been developed for a number of existing plant systems using the design basis document reconstitution process. These documents contain the system design bases which have been extracted from original design documents, previous modification documents, engineering and licensing correspondence. Refer to Attachment 7 for details.

Level of Reviews:
SDDs and MDDs receive an interdisciplinary review, engineering section manager review for technical content, accuracy and comment resolution, and project manager approval. For the development of new SDD's or MDD's, the interdisciplinary review is done either by the Preliminary Engineering Design Review (PEDR) or by the Project Team established for the modification. These

reviews are discussed in detail in Sections 5.8 and 5.9 of this attachment. For the development of SDBD's, the review process is discussed in Attachment 7.

5.4 Core Management

Scope:
Nuclear Fuels is responsible for performing the core designs and reload safety analyses. The reload methodologies employed have been reviewed and approved for application to Oyster Creek by the NRC. The configuration control process employed to provide consistency between the core configuration and performance and the design bases involves four aspects: reload methodology, plant modifications, core loading and core monitoring.

Description:
The first aspect of configuration control for the core is the control of the methodology used to perform the reload analysis. Vendor analyses are covered by their 10 CFR 50 Appendix B Quality Assurance programs. Each analysis that is performed by GPU Nuclear is proceduralized in Fuel Standards. These standards provide step by step instructions for performing the analysis to maintain a consistent application of the NRC approved methodology. The results of the analyses are compared to acceptance criteria to provide compliance with design basis. If results fail to meet licensing criteria, then the reload must either be altered or necessary technical specification changes obtained. The analyses are documented by a calculation report (5000-ADM-7311.01) and independently design verified (5000-ADM-7311.02). The computer codes used in these analyses are maintained under configuration control by procedure (5000-ADM-7340.01) to ensure consistency with approved methodology. Revisions to the fuel standards and computer codes are controlled under the GPU Nuclear OQA Plan. Fuel vendors notify GPU Nuclear of changes in their methods, fuel designs as well as error reporting to allow for evaluation of changes for application to reload methods and design bases. GPU Nuclear routinely performs audits of the reload analysis process and of the fuel vendor.

Plant modifications, including procurement of core consumable (fuel, fuel channels, burnable poison assemblies, control rods, and nuclear instrumentation), are evaluated for their potential impact on reload methodology and design bases. EP-005 identifies those systems that impact reactivity management and require reviews by Nuclear Fuels. New core consumables have a fuel standard (FS5-100) that identifies the procurement process. An additional fuel standard (FS1-104) exists for new fuel designs to identify design basis considerations that must be addressed. Plant modifications and core consumable procurements are performed under the engineering safety review process (5000-ADM-1291.01) which provides the mechanism to update the design bases (FSAR, Technical Specifications, etc.)

as appropriate. Changes resulting from other than formal modifications such as equipment failures and degraded system performance are captured and reviewed for impact on reload analyses through the deviation report, License Event Report and 10 CFR 21 notification processes.

The third aspect of configuration control is to provide consistency between the final core loading configuration and the design loading. This activity provides consistency between the analyzed core loading (FS1-102) and the "as loaded" core configuration (FS1-105).

The final aspect of configuration control is for core monitoring. Shift Engineering is currently responsible for fulfilling reactor engineering requirements. These responsibilities include reactivity management, power maneuvering support and analysis, and low power physics testing. Plant procedures control these activities for compliance with the applicable technical specifications and operating requirements provided in the cycle specific reload design and safety analysis report. The software updates for process computer programs are initiated and controlled by procedure (5000-ADM-7340.02). Cycle dependent inputs are updated, verified and tested each reload to provide for proper monitoring of core performance and technical specification compliance.

5.5 Setpoint Control and Change

Scope:
The control and changing of setpoints and the basis for the setpoints are addressed by Oyster Creek Administrative Procedure 125.4, 'Administration of Setpoints'.

Description:
Setpoints are maintained by a setpoint database within the work management database (GMS2). This database provides a controlled accumulation of information pertinent to defining the elements and variables of setpoints.

For Nuclear Safety Related and Regulatory Required Components, the requirements of the Engineering Standard 'Instrument Error Calculation and Setpoint Determination' (ES-002), are incorporated into the setpoint change documentation. These requirements provide a systematic methodology for determination of setpoints and allowable values, which include providing sufficient margin between the protective action setpoint and the system protection limits to account for all the inaccuracy inherent in the instrument loop.

A safety determination or evaluation is required under the GPU Nuclear Safety Review Process. Support calculations are developed, as required, using procedure 5000-ADM-7311.01 (EP-006),'Calculations' and a design verification is

performed (for NSR and RR components) under procedure 5000-ADM-7311.02 (EP-009), 'Design Verification'.

Level of Reviews:
The section manager reviews the change for technical content and accuracy. Additional engineering discipline reviews are conducted, as applicable, and the PRG Chairman assigns additional reviews as required to verify technical adequacy and to review the potential to affect nuclear safety or safe plant operations. Also, the Safety Review Process provides an RTR and if applicable, an ISR.

5.6 Equipment Control (Switching And Tagging)

Scope:
Oyster Creek Technical Specifications require written procedures be established, implemented and maintained as recommended in Appendix "A" of Regulatory Guide 1.33. Appendix "A" Administrative Procedures require Equipment Controls (Switching and Tagging). The OQA Plan commits to Regulatory Guide 1.33 Rev. 2. Oyster Creek also is subject to the requirements of Occupational Safety and Health Administration (OSHA) regulations (29 CFR 1910). The site specific procedures which implement these requirements are procedures 108 "Equipment Control", 108.5 "Control of Locked Valves and Breakers:, 108.7 "Lockout/Tagout Procedure" and 108.9 "Equipment Alignment and Verification".

Description:
The administrative procedures provide the controls necessary to maintain personnel safety while performing work on plant equipment and that switching operations are performed consistent with equipment protection, nuclear safety concerns and regulatory requirements. All switching and tagging orders within the scope of the Operations Department are prepared by NRC licensed operators and authorized by an NRC licensed Senior Reactor Operator. When authorizing any switching and tagging, the licensed Operations Supervisors are responsible to use their Senior Reactor Operator training, knowledge and experience combined with the administrative procedures to maintain adherence to the plants license requirements. Personnel must meet the training and qualification requirements prior to performing any duties related to Equipment Control. A periodic refresher training is provided and training is also provided when a significant change to the administrative procedure is made.

5.7 Procurement Process

Scope:

Configuration Maintenance provides the technical input required to implement 10 CFR 50 Appendix B requirements and thus provide for the quality of items purchased and installed in safety related applications. A procurement program and dedication process are maintained to maintain the suitability of 10 CFR 50 Appendix B approved vendor material and commercial grade material to meet their intended safety related applications.

Description:
The NRC, through Generic Letters 89-02 and 91-05, and NUMARC, through the Commercial Grade Initiative and the Comprehensive Procurement Initiative, have provided guidance on engineering activities intended to meet 10 CFR 50 Appendix B requirements in an operating plant environment.

Procurement activities are conducted at Oyster Creek based on these generic letters and initiatives, to implement 10 CFR Part 50 Appendix B requirements. These activities include:

- The review of purchase requisitions:

  Configuration Maintenance reviews purchase requisitions to verify adequate identification of the item, specify accurate technical and quality requirements, assure a proper supplier scenario and provide acceptance criteria and methods.

  This practice maintains the design bases of the plant by procuring spare parts to quality requirements of 10 CFR Part 50 Appendix B and technical requirements equal to original supply are specified.

- The classification of component parts:

  Determining the safety classification of components and parts is based upon the regulatory definition of safety-related. It uses a "top down" approach and uses an evaluation of the higher tier levels (system, assembly and component) that was previously performed and reflected in the QCL.

  This activity relates spare parts to the QCL to determine proper safety related procurement. This directs purchase of the item under 10 CFR 50 Appendix B and 10 CFR 21 or as a commercial grade item that will be dedicated.

- The performance of equivalency evaluations:

  The determination of whether a replacement item is "identical" or "alternate" is made. When an item is an alternate, identified differences are evaluated for their effect on the item's function and failure modes. A design characteristics

comparison must conclude that the difference has no adverse affect on the item design function or safety related performance.

This engineering activity provides for the performance of a technical evaluation to ANSI N18.7 (1976) requirements when an original item is not available. This affirms the alternate item meets the design requirement of the original part or component and does not constitute a design change.

- The dedication of Commercial Grade items:

  The dedication process is the combination of activities which establishes whether a Commercial Grade Item is acceptable for use in a Safety-Related application. These activities include identifying failure modes, establishing critical characteristics, and selecting the method of verifying critical characteristics, e.g., receipt inspection/tests, commercial grade survey, source inspection, or combination of methods.

  This engineering process is required for commercial grade items before they can be used as a basic component in accordance with 10 CFR 21. This equivalence in quality and safety function to 10 CFR 50 Appendix B maintains design base requirements at a part and component level.

Material Management

Material Management Department maintains databases of warehouse inventory and procurement activities. These databases provide traceability to item installation and reference to technical and quality requirements. Their inclusion of generic design specifications and part numbers data augments configuration control.

Receipt inspection activities provide a check of purchased material that is part of the 10 CFR Part 50 Appendix B process. Requirements for material traceability and vendor qualifications are addressed. Also this activity is a check that items received meet specifications for safety related material.

Programs addressing warehouse control and storage of materials to prevent degradation of items due to conditions during storage meet 10 CFR Part 50 Appendix B and ANSI/ASME N45.2.2 requirements.

5.8 Project Management and Approval Process

Description:

On December 31, 1995, a new process was put in place for the identification, funding approval, development and implementation of Projects. The controlling document which defines this process is the GPU Nuclear 'Project Approval and Management Process' (1000-PLN-7340.00). This P.A.M.P. plan details how the need for a current or future configuration change is developed, how the design inputs are to be selected, how the design outputs are identified and reviewed, how the configuration change is tracked and how each configuration change is closed-out and turned over to the plant.

Functional groups or individuals within a group can identify a deficiency or opportunity for improvement that may be resolved by a change in plant configuration. A solution to the problem or an implementation of the opportunity is developed by a Performance Team comprising members of the functional groups significantly effected by the specific issue.

Design inputs for Configuration Changes are selected from among controlled configuration documents, history of industry experience, ALARA concerns, and the requirements of applicable Codes and standards. The appropriateness and completeness of these inputs is judged by the design reviews mandated by 1000-ADM-7350.05 "Configuration Change".

The outputs which direct and describe a Configuration Change may be technical calculations, procurement specifications, Bills of Material, revised procedures and construction drawings. These outputs are reviewed under the same mechanism employed to review design inputs. Many of the specific procedures governing these outputs are listed in Section 5.9.

Additionally, engineering documents describing the Configuration Change are required to be rolled-up into controlled configuration documentation via an Engineering Change Document, described in section 5.12.

A requirement of the P.A.M.P. Plan is the development and implementation of a Tactical Plan for Configuration Changes (1000-PLN-7340.00 Exhibit 10). This document lists the engineering, contract planning and inservice deliverables. Tracking of the progress of the Configuration Change is achievable by checking the completion of the Schedule Milestones in the Tactical Plan. All turnover deliverables are also identified in the Tactical Plan, allowing complete tracking of the progress of turnover.

Level of Reviews:
As part of the Project Approval and Management Process (1000-PLN-7340.00), There are two separate reviews for each modification by the Project Team. The Preliminary Engineering Design Review Meeting (PEDR), and Operability, Maintainability, Constructability and Testability meeting (OMTC), as defined in the

'Project Reviews' procedure (5000-ADM-7311.03) provide a formal, systematic review of the modification

5.9 Plant Modifications

Scope:
The Configuration Change Procedure 1000-ADM-7350.05 (EMP-002) defines the requirements and provides guidance for the preparation, review and approval of configuration changes performed at Oyster Creek. Temporary Modifications are addressed in section 5.11 of this attachment.

Description:
The configuration change process uses a graded approach which establishes the minimum documentation requirements for simple configuration changes, then provides guidance for what additional documentation is required for more complex configuration changes. All of the applicable engineering documents are referenced in the Configuration Change Document which serves as the master reference document for the configuration change.

The minimum documentation (when applicable) for every configuration change is shown in the following table with the basis for the documentation.

DOCUMENTATION	BASIS	PROCEDURE
Safety Evaluation Determination Review and / or Safety Evaluation	10 CFR 50.59	Corporate Procedure 1000-ADM-1291.01, "Safety Review Process"
Configuration Change Document	Provides all installation specification requirements, applicable codes, standards and regulatory requirements. If Modification Design Descriptions or System Design Descriptions are not required for this modification, all of this information will be included in this document. See EMP-002 for the detailed documentation required in the Configuration Change Document.	Configuration Change Procedure 1000-ADM-7350.05 (EMP-002)
The quality classification of the work to be performed shall be determined by reviewing the Quality Classification List (QCL). For new components or change of classification of existing components complete the QCL checklist	Assurance that material / components being installed meet the established quality requirements for the system, component or structure in which they are being installed	Quality Classification List 5000-ADM-7313.02 (EP-011)
Fire Protection Evaluation	Provides the method and documentation for compliance with 10 CFR 50 Appendix R	5000-ADM-7370.01 (EP-013)
Environmental Qualification Input and Status Form (if applicable)	Assurance that material / components being installed meet the established environmental qualification for the location which they are being installed	Equipment Environmental Qualification 5000-ADM-7317.01 (EP-031)
Environmental Determination Form (if applicable)	Assurance that material / components being installed do not have an adverse impact on the environment	Environmental Determination 1000-ADM-4500.03

Additional documents may be required depending on the requirements of the configuration change. The specific documentation requirements are governed by the Project Approval and Management Process. Below is a list of some of the significant additional procedures utilized.

DOCUMENTATION	BASIS	PROCEDURE
A Verification Plan and Design Verification are required for a configuration change or portions of a configuration change which are within the scope of the Operational QA Plan for design control	Assurance that the design will perform its intended function consistent with ANSI N45.2.11	Design Verification 5000-ADM-7311.02 (EP-009)
Calculations to support configuration change design (note: calculations for "RR" and "NSR" require verification)	Numerical data to support the configuration change adequacy	Calculations 5000-ADM-7311.01 (EP-006)
Seismic Qualification Methods for USI A-46	Defines the responsibilities, establishes the guidelines and documents the verification of seismic adequacy of components for resolution of USI A-46	5000-ADM-7318.01 (EP-022)
Bills of Material	To determine that the correct material, quality and grade is being installed consistent with the design requirements of the configuration change	Bill of Materials 5000-ADM-6320.01 (EP-028)
Procurement Specifications	To determine that the correct material, quality and grade is being purchased, or fabricated consistent with the design requirements of the configuration change.	Specifications 5000-ADM-7315.01 (EP-004)
Construction Drawings	To provide clear graphic representation of the change for use during installation	GPU Nuclear Drawings 5000-ADM-7312.01 (EP-002)

RR = Regulatory Required
NSR = Nuclear Safety Related

5.10 Alternate Replacements

Scope:
The intent of this process is to provide an abbreviated configuration change process for alternate replacements, which meets all of the requirements necessary for a configuration change. The process at Oyster Creek has been developed to meet the intent of EPRI Guideline NP-6406 - "Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants."

Description:
Alternate replacements are a type of configuration change that is used when a component is replaced with one that is comparable in terms of its design, materials of construction, manufacturing process and quality assurance requirements such that the replacement will perform its intended function as determined by an engineering review. The Configuration Change Procedure - 1000-ADM-7350.05 (EMP-002) is the controlling document for alternative replacements. This procedure allows for this type of replacement to be handled by site specific procedures, providing that they meet the requirements of the Operating Quality Assurance Plan - 1000-PLN-7200.01. The site specific procedures for alternative replacements are 'Control of Engineering Directed Alternate Replacements' (124.2) and Conduct of Engineering - Engineering Evaluations' (125). These procedures define the requirements and provide guidance for the preparation, review and approval of alternative replacements, consistent with the requirements of The Configuration Change Procedure (1000-ADM-7350.05) for the review of engineering configuration concerns. This is done so that replacement components do not alter the design of the system and that applicable configuration control documents are updated and plant operating procedures adequately reflect the plant configuration. Specific elements of these procedures are as follows:

- Critical characteristics for design of the replacement component are selected.

- The item's functional quality classification is determined and if Nuclear Safety Related (NSR), how its function related to the host component.

- A detailed engineering evaluation is conducted comparing the critical characteristics for design of the original and replacement component.

- A safety determination / evaluation is performed in accordance with the site procedure Nuclear Safety Review (130) which is consistent with the safety review process procedure 1000-ADM-1291.01 described above.

- Plant Procedures, technical manuals, preventive maintenance, etc. are reviewed and revised if required, consistent with the controlling modification procedure.

- Configuration control documents are revised upon component installation, consistent with the controlling modification procedure.

Level of Reviews:
Alternate replacements are reviewed and approved by both the cognizant engineer and the engineering manager. Quality Verification review and approval are required for alternative replacements for RR and NSR components (and when welding is required) which fall within the Operational Quality Assurance Plan.

5.11 Temporary Modifications

Scope:
The intent of the temporary modification process is to provide an expedient method to make temporary plant configuration changes that support ongoing plant evolutions and meet the requirements of 10 CFR 50.59 and the Operational Quality Assurance Plan.

Description:
The controlling document for temporary modifications is 'Temporary Modification Control' station procedure 108.8.

Safety Review:
Oyster Creek procedures contain the requirement that a safety evaluation / review be conducted consistent with 10 CFR 50.59 and the documentation of the review is integrated into the site specific procedures. Temporary modifications use the corporate safety review process, as previously described in this letter, but add additional requirements as follows. A safety determination is performed to assess if a safety evaluation is required. If a safety evaluation is determined not to be required, then the qualified Responsible Technical Reviewer signoff is sufficient, unless the Temporary Modification affects a critical component or system, then a second RTR review and signoff is required on either the safety determination and / or the safety evaluation. The second RTR must be independent of the first and both can not be assigned to the same shift in the control room. A qualified Independent Safety Review is required of the safety evaluation (if one is required), but the implementation of the temporary modification can proceed with the ISR as a follow-up action.

Design Review:
Oyster Creek procedures require that a technical / design review be performed, and documented by the Engineering Division. The reviews address technical

adequacy of the temporary modification consistent with the engineering configuration attributes listed in the Configuration Change Procedure (EMP-002) which are used to plant modifications.

Plant configuration documentation required to reflect the temporary modification (such as marked up drawings, procedure changes) is required to be in place to support the installation.

The procedures establish the requirement for regular review of each installed temporary modification to assure that the basis, need and plant condition are still applicable to the temporary modification installation and the documentation that supports it.

Level of Reviews:
Each temporary modification must be reviewed and approved, at a minimum, by the shift manager and a qualified RTR.

5.12 Engineering Change Documents

Scope:
During the course of operations, maintenance and plant walkdowns, physical plant configurations are identified which are not consistent with documentation. These "as-found" conditions must be identified and evaluated so that the physical plant configuration is accurately reflected in the configuration documentation. Additionally, during the course of the development and implementation of a configuration change, alterations in the controlling engineering documentation may have to be made. The "Engineering Change Document", Procedure 5000-ADM-7350.03 (EMP-015), provides a mechanism to serve both needs.

Description:
Once identified, the required changes are written on an ECD form and submitted with the affected documents to the cognizant engineer for disposition. In the case of "as-found" conditions, the cognizant engineer is the System Engineer. If the proposed change or "as-found" condition is acceptable, the engineer provides the disposition of the changes and obtains technical approval from the functional engineering supervisor, project approval from the project manager, and Quality Verification approval if the change affects QV requirements. There are five review requirements which are required to be evaluated: Safety Evaluation, Design Verification, Environmental Qualification, Fire Hazards Analysis and Environmental Determination. Proposed changes or "as-found" condition which are determined to be unacceptable must be removed or modified, and the ECD is not approved. Approval of the change is complete when the appropriate reviews are completed. Completed ECDs are posted in the Electronic Data Management

System against the effected documents to maintain engineering document configuration control.

Level of Reviews:
The reviews associated with an ECD are intended to be consistent with those of the original document against which the ECD is to be posted. In the case of "as-found" conditions, these reviews must address technical adequacy and consistency with FSAR and system design requirements. In either case if a safety review requirement is established the document is subject to the requirements of the Safety Review Process.

5.13 Start up and Test Program

Scope:
Post installation testing of plant modifications to confirm the plant , as modified, functions in accordance with design.

Description:
Modification testing consists of component level and system level tests. A graded approach is used in developing test requirements such that complex or safety related modifications generally receive a greater extent of testing than simple or balance of plant modifications. Components affected by modification are tested using generic procedures which are applied to all like components. Upon the completion of component level testing integrated system level testing is performed. To the extent practicable, these tests confirm the design bases and demonstrate the capability of systems to meet their performance requirements including abnormal operating and failure modes. Test requirements are developed jointly by the test engineer and the modification design engineer considering performance requirements and design bases.

Level of Reviews:
Modification test requirements and functional test procedures are given multidiscipline reviews and approved by the Test Approval Group (TAG). The "Safety Review Process " procedure 1000-ADM-1291.01 is applied to functional test procedures and functional test procedure changes. As a minimum, TAG consists of a representative of the test organization, a representative from design engineering and a representative from plant operations. The plant operations representative must hold or have held a reactor operator license or certification. TAG approval indicates agreement with the technical content of the testing to be performed, including items such as , a scope of testing, inclusion of test requirements necessary for design verification, test methods and acceptance criteria. Each TAG member solicits additional review from within the organization to fulfill these requirements.

5.14 FSAR Updates

Scope:
The method for the control of changes and revision of the Updated FSARs (UFSAR) for TMI-1 and Oyster Creek is specified in Corporate Procedure No. 1000-ADM-7320.0l, "Update FSAR Document Change Control". This procedure establishes the documentation requirements and the level of reviews required to control changes to the FSAR.

Description:
Changes to the FSAR are originated as part of the "Safety Review Process" described in Corporate Procedure No. 1000-ADM-1291.01 as identified in safety evaluations generated in accordance with 10 CFR 50.59. Safety evaluations are generated for plant modifications and/or procedure change requests and/or tests and experiments, or, in the event errors in the FSAR are discovered. When a change or revision is initiated, the change and the basis for the change, receives applicable reviews, such as an interdepartmental review, Responsible Technical Review, Independent Safety Review and licensing review. Changes are then compiled for each FSAR chapter and reviewed by an Assigned Chapter Review Coordinator who is selected based on his/her area of technical expertise. The compiled changes are then published biennally as a revision to the FSAR and submitted to NRC in accordance with 10 CFR 50.71(e). The submittal of the FSAR revision is reviewed by applicable department heads and managers.. The latest update of the Oyster Creek FSAR is Update 9, which was submitted May 19, 1995.

5.15 Plant Procedure Development and Change

Scope:
Development of new procedures or changes to site specific procedures , used to operate and maintain Oyster Creek, are addressed in station procedure 107, 'Procedure Control' which is used in conjunction with procedure 103, 'Station Document Control'. Additionally, procedure 107 provides the methodology for executing temporary changes to procedures.

Description:
The procedure development and change process (including temporary changes) implements the requirements of 10 CFR 50.59 by using the safety review process as described above. Biennial reviews are conducted on all station procedures, consistent with the OQA Plan, by an individual knowledgeable in the area affected by the procedure to determine if changes are necessary or desirable.

Level of Review:
Procedures and changes to procedures are prepared by an individual knowledgeable in the affected area. Each new procedure or procedure change is subject to the Safety Review Process as described in Section 5.1.

5.16 Engineering Procedure Development and Change

Scope:
Engineering procedures provide instructions to personnel for meeting corporate and divisional requirements and to control activities necessary to accomplish work. Engineering Department personnel are required to adhere to these procedures so that their work will be performed in accordance with the requirements of the GPU Nuclear Operational Quality Assurance (OQA) Plan

Description:
Engineering division procedure 5000-ADM-1218.01 (Technical Functions Division Policy and Procedures) provides direction on the format requirements and the review, approval, and revision process for engineering department procedures. Engineering department procedures are used to provide specific direction on the conduct of critical activities and processes. Examples of these include but are not limited to administrative processes, calculation control, safety review, design verification, and configuration control activities. Adherence to these procedures maintains the design, configuration, and operation of the plant within its design bases.

Level of Review:
New and revised engineering procedures will be reviewed and approved by engineering department management with additional review by interfacing division personnel as appropriate. All procedures must indicate if the subject is within the scope of the GPU Nuclear OQA Plan and whether or not the new procedure or substantive procedure change requires a safety and review. OQA Plan, Section 2.0, is used to determine if the procedure is in the OQA Plan scope. Safety reviews are conducted in accordance with engineering procedure 5000-ADM-1291.01, "Nuclear Safety/Environmental Determination and Evaluation". Biennial reviews are performed in accordance with procedure 1000-ADM-1218.01, "GPU Nuclear Corporate Policies, Plans, and Procedures", to provide a periodic evaluation of procedure content.

5.17 Control of Drawings

Description:

The preparation, review, and approval of drawings prepared by in-house designers or by design organizations under contract are controlled via the requirements of 5000-ADM-7312.01 "GPU Nuclear Drawings. This procedure also establishes control for revision of existing drawings to incorporate changes in engineering or design. Revision to existing drawings to document as-found or as-constructed plant conditions are controlled by 5000-ADM-7312.02 "Use Drawings".

These procedures establish a system of checks and oversight for the approval of new and revised drawings. Release of completed drawings is via the Configuration Management Transmittal (1000-ADM-1215.02). Once released, drawings are controlled within the GPU Nuclear Electronic Document Management System. This system controls the transfer of hard copies for revision by a system of transmittal slips. Drawings existing as an electronic file (CAD drawings) are revised electronically and issued to Information Resource Management as a replacement to the original file, subject to the revision rules of 5000-ADM-7312.01.

5.18 Control of Calculations

Description:
The documentation and control of manual and computer calculations is governed by the requirements of 5000-ADM-7311.01 (EP-006), "Calculations". This procedure covers requirements on both stand-alone calculations and calculations that are included as part of another document.

The general format requirements for calculations are set forth in the above procedure, as well as the schedule of approvals and verifications required to release a calculation for use. Calculations are to be checked by the responsible section manager; if verification is required, the section manager selects a verification engineer and the requirements of 5000-ADM-7311.02 (EP-009), "Design Verification", apply.

Checked and verified (as necessary) calculations are released for use via a Configuration Management Transmittal (1000-ADM-1215.02) and entered into the Electronic Document Management System for retention and distribution. In the event that a revision to the calculation is required, procedural controls are established in EP-006.

Attachment 2

Oyster Creek Data Analysis Team Documents Reviewed

The following table lists documents reviewed by the Oyster Creek Data Analysis Team. Data from these documents was evaluated by the team to assess design bases conformance, configuration control, and corrective action effectiveness.

Oyster Creek Data Analysis Team Documents Reviewed

Title Of Document / Information Reviewed
1990 Licensee Event Reports
1991 Licensee Event Reports
1992 Licensee Event Reports
1993 Licensee Event Reports
1994 Licensee Event Reports
1995 Licensee Event Reports
1996 Licensee Event Reports
Audit F-OC-96-01, Fuel Assembly Manufacture
Audit F-OC-96-02, 16R Fuel Related QA
Audit F-OC-96-03, Fuel Assembly Records
Audit F-OC-96-04, Fuel Channels
Audit O-COM-94-16, Corrective Action
Audit O-COM-94-35, Corrective Action
Audit O-COM-95-02, Corrective Action
Audit O-COM-95-06, Corrective Action
Audit O-COM-95-09, Tech Support & Design Control
Audit O-OC-95-01, Plant Process Computers
Audit O-OC-95-05, Fuel Management
Audit O-OC-95-06, Core Performance
Audit O-OC-96-01, Design Control
Audit S-OC-95-02, Plant Engineering
Audit S-OC-95-06, Plant Operations
Audit S-OC-95-09, Fire Protection
Audit S-OC-95-10, Safety Review Program
Audit S-OC-95-12, Maintenance
Audit S-OC-96-05, Plant Engineering
Audit S-OC-96-08, Fire Protection
Audit S-OC-96-14, Maintenance
Design Basis Information Concerning SEP & FTOL Evaluation
Deviation Reports 1995-1996 (Pre-sorted DRs associated with Design Basis, Configuration Control, Modifications, FSAR) See Note (2)
GPU Nuclear Review of Processes
NRC Diagnostic Evaluation Team Inspection
NRC IR 88-202 SSOMI (Design) NRC IR 88-203 SSOMI (Installation & Test)
NRC IR 89-80 ESW/CS SSFI
NRC IR 93-81 OSTI
NRC IR 94-22

Title Of Document / Information Reviewed
NRC IR 94-23
NRC IR 94-24
NRC IR 94-25
NRC IR 94-26
NRC IR 94-27
NRC IR 94-29
NRC IR 95-01
NRC IR 95-02
NRC IR 95-05
NRC IR 95-06
NRC IR 95-07
NRC IR 95-08
NRC IR 95-09
NRC IR 95-11
NRC IR 95-12
NRC IR 95-16
NRC IR 95-22
NRC IR 95-23
NRC IR 95-24
NRC IR 95-99 SALP 12/93 - 6/95
NRC IR 96-01
NRC IR 96-02
NRC IR 96-03
NRC IR 96-05
NRC IR 96-06
NRC IR 96-07
NSA Assessment 96-11 FSAR Technical & Conformance Review
NSA Monitoring 9512009, Equipment Control Restoration
NSA Monitoring 9521002, EQ Component Maint.
NSA Monitoring 9521009, Document Control
NSA Monitoring 9523001, Fire Protection Walkdown
NSA Monitoring 9523002, Fire Protection Walkdown
NSA Monitoring 9612003, Equipment Control Review
NSA Monitoring 9621003, Trunnion Room Fan Mod
SDBD-OC-212 Automatic Depressurization
SDBD-OC-212a Core Spray
SDBD-OC-220 Reactor Coolant
SDBD-OC-225 Control Rod Drive
SDBD-OC-241 Containment Spray
SDBD-OC-421 Condensate & Feedwater
SDBD-OC-532 Emergency Service Water

Title Of Document / Information Reviewed
SDBD-OC-535 Circulating Water
SDBD-OC-641 Reactor Protection
SDBD-OC-661 Radiation Monitoring
SDBD-OC-740 Emergency Power
SDBD-OC-822 Standby Gas Treatment
SDBD-OC-838 Drywell Cooling
SDBD-OC-852 Compressed Air
TDR 986 Electrical Distribution SSFI
TDR 1002 ADS SSFI
TDR 1028 Core Spray SSFI
TDR 1118 RPS SSFI
TDR 1176 ESW SWSOPI

Notes: (1) The following NRC Inspection Reports were screened and found not to contain any usable data for the 10 CFR 50.54(f) response.

NRC IR 95-03
NRC IR 95-04
NRC IR 95-10
NRC IR 95-14
NRC IR 95-19
NRC IR 95-20
NRC IR 95-21 Duplicate of SWOPSI
NRC IR 96-04
NRC IR 96-08

(2) 40 Individual Deviation Reports were reviewed.

(3) 24 Individual LERs were reviewed covering the period 1990-1996.

Attachment 3

GPU Nuclear SSFIs and Self Assessment

This attachment provides details on the following SSFI's and Self Assessment:

1. Oyster Creek Emergency Electrical Power Distribution SSFI - TDR No. 986

2. Oyster Creek Automatic Depressurization System SSFI - TDR No. 1002

3. Oyster Creek Core Spray System SSFI - TDR. No. 1028

4. Oyster Creek Reactor Protection System SSFI - TDR No. 1118

5. GPU Nuclear Self Assessment - Service Water System Operational Performance Inspection

GPU Nuclear SSFIs and Self Assessment

This attachment provides a summary of the results of four Safety System Functional Inspections and one Operational Performance Inspection performed by GPU Nuclear. The objective is to provide a detailed sense of the breadth and depth of these "vertical slice" reviews.

The summaries of the SSFIs provided are those provided by the executive summaries of the associated Technical Data Reports (TDRs) with some editorial changes. Of significance are the following:

- the results are presented as they were reported; not as finally evaluated and resolved;
- all the specific "open items" that were reported for further evaluation are not included.
- two of the four SSFIs were performed during the "pilot program" period (i.e., 1989) prior to any significant design basis reconstitution effort being completed as described in Attachment 7.

1. Oyster Creek Emergency Electrical Power Distribution SSFI - TDR No. 986 (August, 1989)

The inspection results are summarized with reference to the GPU Nuclear SSFI Specification (SP-9000-56-009) General Requirements.

- It could not be determined that GPU Nuclear had defined, beyond the scope of the FSAR, which documents specifically describe the design basis. Documents, which contained design basis information, were identified. In those documents, certain inconsistencies, and in some instances, a lack of documentation to support the design basis was identified. It was recommended that a reconstitution of the design basis be performed.
- The emergency electrical system was found to meet the requirements of the design basis accident.
- A lack of central control of design basis documentation was found, which made it difficult to assure that the design basis would not be compromised when modifications were made to the emergency electrical distribution system.
- The quality of the operating procedures as well as the overall conduct of routine operations, within the plant was considered to be a strength.

- System testing to be adequate although more in depth trending of conditions was recommended. While the team considered maintenance was adequate, improvements were recommended in the administrative control of maintenance and in the structure of the preventive maintenance program.

- The plant operators were found to be knowledgeable in the electrical distribution system operation under abnormal conditions.

- The team found inconsistencies in licensing basis documentation. As noted above, a reconstitution of the system's design basis was recommended; this would also enhance the licensing basis documentation.

15 open items and two Quality Deficiency Reports were issued for further evaluation. These items were:

- a review of short circuit calculations and relay studies should be performed to determine if relay settings for emergency buses should reflect conditions of minimum short circuits occurring;

- a review of a safety classification undervoltage trip devices, installed to reduce diesel generator loading, should be performed with the review towards not taking credit for load shedding unless the devices were classified as nuclear safety related;

- a review of the coversheets of Startup and Test turnover packages to ensure that references for the procedures used were listed;

- an evaluation of changes made to SCEW and SCEW sheets should be performed to ensure environmental qualification of electrical equipment was maintained;

- an evaluation of cable labeling should be performed to determine if new labels were required;

- an evaluation of the need for a revision to surveillance test procedure, 636.2.001, to avoid "pre-conditioning" breaker trip devices;

- an accessible organized list of supporting design information with appropriate cross references should be provided;

- an evaluation of the Cable Trending Program should be performed to determine if sufficient testing was being performed;

- an evaluation of housekeeping discrepancies should be performed;

- a review of a paragraph of procedure 337 should be performed with the view toward clarifying the phraseology;

- a DC voltage drop study should be performed to ensure that sufficient voltage would be supplied to the loads required to operate;

- a revision to FSAR paragraph 8.2.1 should be made to clarify when fast bus transfers will not occur;

- a review of original construction cable pull schedules should be conducted with the view towards incorporating them into the GPU Nuclear record system;

- a change to three drawings (P&IDs) should be initiated to have them designated as "Nuclear Safety Related"; and,

- a review of the process of maintaining calculations needs to be performed to ensure that load changes are actually incorporated.

All these open items and Quality Deficiency Reports have been evaluated and resolved. Note, that some of the items may have been evaluated as requiring no further action.

The two Quality Deficiency Reports address:

- The need to evaluate discrepancies in controls of preventative maintenance, including the need to provide guidance for skipping maintenance items.

- The need to evaluate the impact of discrepancies in plant procedures 105, 107, 118.1, and GMS-2, superseding of revisions, performance of failure analyses, etc.

All these open items and Quality Deficiency Reports have been evaluated and resolved. Note, that some of the items may have been evaluated as requiring no further action.

It should be noted that this SSFI was conducted prior to completing a reconstitution of this system as described in Attachment 7 of this letter.

2. Oyster Creek Automatic Depressurization System SSFI - TDR No. 1002 (September, 1989)

The overall objective was to assess whether the Automatic Depressurization System's (ADS) bases and functional capacity had been modified, operated, maintained, and tested in a manner that would lead to the conclusion that the system would perform as designed in an accident or transient.

The governing document for the inspection was Specification SP-9000-56-009, "Technical Specification for Safety System Functional Inspection for Three Mile Island Unit 1 and Oyster Creek Nuclear Generating Station." The overall method for the inspection was to:

- research design bases information for the system;
- research the work methods and practices which impact the design bases of the system (e.g., modifications, operations, maintenance, testing, procurement, and training); and,
- review and assess the design bases, processes, procedures and finished products potentially affecting implementation of the design bases of the system.

This approach provided the overall structure for collecting, reviewing and assessing information. The details of the review and the overall format of the report followed the guidance contained in the GPU Nuclear specification. Walkdowns, interviews and document review were utilized during the inspection.

The overall conclusion reached was that the Automatic Depressurization System is expected to function as designed, per the currently intended bases, in an accident or transient. There have been several major re-evaluations that required thorough review of the system since Oyster Creek Nuclear Generating Station began commercial operation. These reviews have verified the design bases for the system. These evaluations included reload analyses, the safe shutdown analysis for 10 CFR 50, Appendix R, the Probabilistic Risk assessment, the Mark I Long Term Containment Program, the Systematic Evaluation Program, NUREG 0737 and its Supplement 1, the implementation of the General Electric SAFER/CORECOOL/GESTR transient analysis codes and the Environmental Qualification Program.

Several strengths were identified during the inspection.

- Training for operators and maintenance crafts were assessed as complete and strongly based on the practical aspects of the trainee's job functions. The confidence and enthusiasm demonstrated by the people who were interviewed was particularly noteworthy.

- The post-modification test program was well-integrated into the modification process and testing criteria corresponds to equipment function and functional performance requirements.

- The procedures which govern operations and the administration of the Surveillance Test Program were clear, complete and accomplished their objectives. The acceptance criteria in the Surveillance Test Program for recalibration of instruments minimizes the potential for out of specification as-found conditions on the next surveillance. This enhances system reliability and increases the confidence in system performance during the time interval between surveillances.

Several areas where improvements would enhance confidence in system performance were identified.

- Design and licensing bases information were difficult to retrieve. It is difficult to know when sufficient information has been obtained to form the starting point for modifications, 10 CFR 50.59 evaluations and other design-related evaluations.

- The power transfer function for the Electromatic Relief Valves was not being tested. This function was necessary to assure the availability of three valves with a single failure.

- Several circuit paths in the system logic were not being tested during the logic functional test. The complexity of the logic places a high reliance on testing to assure proper function.

- Test results on the Electromatic Relief Valves indicated that their measured capacity was less than their nominal 600,000 lbm/hr flow rate. The valves appeared to be conservatively modeled in the Reload Analysis in which they were shown to have sufficient capacity to protect the reactor coolant system, however their expected flow rate under design conditions should have been documented. This would prevent the use of nonconservative values if the analytical model were revised in the future.

Thirty-two open items, which included two preliminary safety concerns were reported for further evaluation. These 32 items have been evaluated and resolved. Note that some of the items may have been evaluated as requiring no further action.

It should be noted that this SSFI was conducted prior to completing a design bases reconstitution for this system, as described in Attachment 7 of this letter.

3. Oyster Creek Core Spray System SSFI - TDR. No. 1028 (August, 1990)

The overall objective was to assess whether the system has been modified, operated, maintained and tested in a manner that would lead to the conclusion that the system would perform as designed during an accident or transient.

The governing document for this inspection was GPU Nuclear Specification SP-9000-56-009, Rev 1, "Technical Specification for Safety System Functional Inspection of Three Mile Island Unit 1 and Oyster Creek Nuclear Generating Station." The overall method for the inspection was to:

- research design bases information for the system;

- research the work methods and practices which impact the design bases of the system (e.g., modifications, operations, maintenance, testing, procurement, and training); and,

- review and assess the design bases, processes, procedures and finished products potentially affecting implementation of the design bases of the system.

This approach provided the overall structure for collecting, reviewing and assessing information. Walkdowns, interviews and document review were utilized during the inspection.

There have been several major evaluations that required thorough review of the system since Oyster Creek Nuclear Generating Station began commercial operation. These reviews included a major modification that was installed in 1976 that appeared to define applicable codes and standards better than most modifications performed during that time period. Other reviews were performed pertaining to the Power Uprate, Systematic Evaluation Program, NUREG 0737, Environmental Qualification Program, and Appendix R to 10 CFR 50.

Several strengths were identified during the inspection.

- The plant labeling program had been comprehensively implemented.

- Craft training integrated classroom training with on-the-job training. It also incorporated plant-specific and industry operating experience.

- The Failure Modes and Effects Analysis performed by Fluor-Daniel was viewed by the team as a strong initiative taken by GPU Nuclear in response to a problem found while engineering a modification.

- Operator training was performed in a timely and effective manner on modifications.

- Potentially generic NSSS operating experience documents were shared between TMI-1 and OCNGS.

- Surveillance test procedures were clearly presented and easy to use with a level of detail that aids in proper performance of the tests.

- Operator aids were carefully controlled and tracked.

Several areas where improvements would enhance confidence in system performance were identified.

- Design and licensing basis information was difficult to retrieve. This may have contributed to some of the inconsistencies that were noted by the inspection team in technical evaluations performed at different times on the same portion of the system.

- The water hammer issue was reopened by GPU Nuclear shortly before the SSFI began. The system had been experiencing water hammer incidents since initial plant operation. There was a long history of evaluations of the problem and corrective actions had lessened the severity; however, the problem persisted.

- The Core Spray System relief valves had a reset pressure that is close to the shutoff head of the pumps. One of the relief valves stuck open during a surveillance during the last week of the SSFI.

- Testing performed on a Core Spray Pump 5KV cable, in response to the latest failure that was documented in LER 90-05, was less comprehensive than testing performed on other 5KV cables. The testing that was performed utilized a lower voltage than other tests.

- There were several observations written on the completeness and consistency of structural analysis of pipe stress.

The overall conclusion was that the Core Spray System was expected to function as designed during accident or transient conditions. This overall conclusion was supplemented by the determination that GPU Nuclear continued to improve implementation of its established programs and supporting procedures.

46 open items were issued for further evaluation. These open items included one preliminary safety concern and three Material Nonconformance Reports. Note that some of these items may have been evaluated as not requiring any further action; or, were associated with ongoing programs (e.g., SQUG, GL 89-10, etc.)

Of these 46 open items, two remain open. Both are associated with evaluating the harsh environment accuracy of the RE-17 pressure switches (the correctness and consistency of the analytical bases of plant setpoints).

It should be noted that this SSFI was conducted prior to completing a design bases reconstitution, as described in Attachment 7 of this letter.

4. Oyster Creek Reactor Protection System SSFI - TDR No. 1118 (May, 1993)

The primary objective was to determine if the system had been modified, operated, maintained, and tested in a manner that was consistent with its design basis. A secondary purpose was to verify and validate design basis information contained in the RPS Design Basis Document (SDBD-OC-641, Rev. 1).

The governing document for this inspection was GPU Nuclear Specification No. SP-9000-56-009, Rev. 4, "Technical Specification for Safety System Functional Inspection of Three Mile Island Unit 1 and Oyster Creek Nuclear Generating Station." The inspection method consisted of:

- Reviewing and evaluating the system design basis information presented in SDBD-OC-641 (Rev. 1), "System Design Basis Document For Reactor Protection System."
- Evaluating the various programs and procedures that are used to verify, maintain or preserve the design basis of the system (e.g., engineering, modifications, procurement, operations, maintenance, testing, and training).
- Performing system walkdowns and conducting personnel interviews.

The following strengths were identified during this inspection:

- Electrical schematics and other applicable engineering drawings were determined to be accurate and maintained up-to-date such that they correctly reflect the "as-built" configuration of the RPS.
- Surveillance test procedures were determined to be technically sound, accurate, and complete with one exception. This exception is applicable to time delay functional testing and is discussed in Observation No. 641-10.
- Based on a review of procurement activities applicable to several RPS components, Plant Engineering's Commercial Grade Item Dedication Program was determined to be technically sound and appropriately implemented.
- Identification of RPS components, with the exception of fuses, was found to be both comprehensive and accurate during system walkdowns.
- The System Design Basis Document was determined to be accurate with several minor exceptions. Its full benefit will be realized when it is updated to incorporate supplemental design documentation as it is identified. Recommendations to correct or enhance this document are presented in various inspection observations.

- The implementation of training program elements applicable to licensed operator requalification, maintenance technician, and I&C technician training was characterized as a strength. This conclusion was based on a limited scope of review and evaluation of GPU Nuclear's overall training program, as applicable to the RPS.

The inspection team identified (3) adverse conditions that potentially diminish the reliability of the RPS. They pertained to the single-failure criteria and are as follows:

- The RPS Instrumentation Upgrade Modification, implemented in 1986, did not adequately consider RPS logic circuit separation with respect to the installation of the scram pilot valve load group status lights on Panel 4F.

- Several concerns regarding the maintenance of RPS circuit separation were identified during system walkdowns.

- Based on a review of RPS connection diagrams and visual inspections of RPS Panels 6R and 7R, the inspection team could not determine if the RPS 120 V Vital AC system was adequately grounded. Failure to establish and maintain proper grounding could cause a spurious scram or prevent the scram function from occurring if actually challenged during a design basis accident.

The implementation of the Environmental Qualification (EQ) Program was evaluated. The methods used to maintain EQ requirements during scheduled preventive maintenance component replacement activities were evaluated by reviewing completed EQ Job Order Packages. Based on a limited review of only those RPS components contained within the scope of the EQ Program, it was concluded that EQ Program implementation constituted a potential weakness. Earlier in 1993, the EQ Group had requested that the QA Department conduct an audit applicable to the implementation of the EQ Program. Based on EQ Program implementation issues identified during the SSFI, it was concluded that a performance based QA audit of the methods used to implement the OCNGS EQ Program was warranted. The methodology used by GPU Nuclear to determine RPS setpoints was evaluated by comparing the methodology established in GPU Nuclear Engineering Standard ES-002 (Rev. 3), "Instrument Error Calculation and Setpoint Determination", with that of the latest revision of ISA Standard S67.04, the current recognized industry standard. A technical review of each existing RPS setpoint was also performed. The inspection team used the applicable elements of NRC Inspection Procedure 93807, "Systems Based Instrumentation and Control Inspection", as a guideline for this review.

Based on the results of the evaluation of Engineering Standard ES-002 and the technical review of RPS setpoint calculations GPU Nuclear's existing setpoint methodology was characterized as a programmatic weakness. A complete setpoint calculation that addresses all potential instrument loop inaccuracies was recommended to be documented as a design basis source of information. There was no existing setpoint calculation applicable to the Recirculation Loop Flow-High trip setpoint. Several of the RPS setpoint calculations reviewed were determined to be incomplete as they did not adequately address all potential instrument loop inaccuracies. Various other RPS setpoint calculations did not provide sufficient reference to design/technical data sources or adequately document assumptions used in the calculations. However, it was noted that GPU Nuclear had appropriately analyzed the performance of 39 instruments addressed by the plant's Technical Specifications with respect to instrument drift. As a result of that analysis, several instruments were replaced with upgraded instruments that exhibit improved performance. Prior to this SSFI, GPU Nuclear had contracted GAI to perform an assessment of its setpoint methodology as established in Engineering Standard ES-002, Rev. 3, as well as to evaluate the technical adequacy of several setpoint calculations. The results of this assessment were reviewed and found in general, to parallel the observations documented by the inspection team. No case where a trip setpoint was inappropriately determined was identified.

GPU Nuclear's efforts to develop technically accurate and complete System Design Basis Documents, and their verification by the SSFI process, was considered a significant strength. The setpoint methodology/program was the only apparent weakness in GPU Nuclear's overall System Design Basis Document Program.

The overall conclusion reached was that the RPS has been maintained in a configuration consistent with it's design basis and was expected to function as designed during a design basis

accident. GPU Nuclear performed preliminary evaluations to determine the safety significance of each observation.

Fifty-three open items were reported for further evaluations. Note that some of these items may have been evaluated as requiring no further action. Of these 53 items seven remain open. These seven are related to:

- originating change requests for the FSAR and the RPS system design basis document (SDBD-OC-640);
- setpoints and tolerances; and
- revision to an engineering standard related to determining setpoints.

5. GPU Nuclear Self Assessment - Service Water System Operational Performance Inspection (October, 1995)

This self assessment inspection was performed by GPU Nuclear and contractor personnel and was conducted in accordance with the guidelines of NRC Temporary Instruction (TI) 2515/118, Revision 1, "Service Water System Operational Performance Inspection (SWSOPI)." The reduced scope inspection plan was reviewed by NRC and concurred with in NRC letter to GPU Nuclear dated October 18, 1995. Although the initial reduced scope plan was to account for the NRC Safety System Functional Inspection performed in 1989 on the containment spray and emergency service water systems, the actual inspection essentially covered all elements of the TI. The self assessment also reviewed the follow-up actions that resulted from the NRC SSFI findings.

The following objectives of a full scope SWSOPI were fulfilled:

- Assess planned or completed actions in response to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment,"

- Verify the ESW system was capable of fulfilling it's thermal and hydraulic performance requirements and was operated consistent with its design bases, and

- Assess the operational controls, maintenance, surveillance and other testing, and personnel training to ensure the ESW system was operated and maintained so as to perform it's safety related functions.

This assessment was accomplished by performing a review of the OCNGS ESW system components and system performance including design requirements, operation, maintenance, surveillance and other testing practices; maintenance and performance history; quality assurance and implementation of corrective actions. The assessment also reviewed the ESW heat exchanger performance testing program, piping inspection program, blue mussel monitoring program, and the safety review process as applied to ESW system activities.

The design review assessed the technical adequacy of the ESW system concentrating on essential safety and functional characteristics. The scope of design review considered design conditions and transients, component classification, equipment qualification, single failure criteria, potential flooding, common mode failure, corrosion/erosion due to flow and biological mechanisms, and a selection of other attributes that contribute to the effectiveness of the system. Engineering calculations and analyses were found to appropriately establish or verify ESW system design basis requirements, and that these requirements had been accurately reflected into the system's applicable surveillance test procedures. The timeliness of updates to design basis documents including calculations and the FSAR were identified as areas needing process improvements. These issues have been addressed. Overall, the inspection team determined that engineering and design

activities had performed to the extent required to ensure that the design basis of the ESW system was preserved.

The operations review consisted primarily of a review of system procedures, emergency operating procedures, alarm response procedures, and standing orders. Operator training program material was reviewed to assess the level of detail provided in the system's design, safety functions, and operation methods. Inclusion of modifications in the training program was also evaluated. Operational controls that ensure proper valve alignments of functioning components was assessed. The availability of essential system status information to the operator was also evaluated. The inspection determined the overall operation of the ESW system to be acceptable. In general, procedures used to operate the ESW and support systems were determined to be adequate with some areas of needed improvement identified. Operations personnel were found to be knowledgeable of the systems and their safety functions. Operator training was found to adequately reflect the current plant configuration.

The maintenance review assessed the adequacy of maintenance programs and procedures to ensure the system would perform it's safety related function. Maintenance records were reviewed to verify that safety-related components were addressed by the maintenance program. Programs and processes utilized for equipment failure trending, corrective actions implementation, and root cause analysis were assessed. The adequacy of post-maintenance testing programs were evaluated. The inspection identified that a more aggressive corrective maintenance approach was needed for some long-standing deficiencies. These items have been addressed. Based on the review performed of the routine maintenance activities, as well as the corrective maintenance history, the inspection concluded overall that maintenance was effective at maintaining overall system availability at a high level.

Surveillance and testing verified that testing performed on the system was sufficient to demonstrate capability to perform the intended functions during the most severe operating conditions. The inspection focused on the functional testing of the system and components. The inspection determined that the In-Service Test (IST) Program was well defined and pump flow and vibration data was found to be effectively trended. Technical Specification surveillance testing evaluated was determined to be adequately proceduralized and implemented.

Quality assurance program activities including corrective actions, operability determinations, trending, and quality verifications were evaluated to determine the adequacy of these programs in identifying and correcting deficiencies. The inspection concluded that quality assurance activities are generally effective providing aggressive and timely response to most issues. Some long standing issues were identified which indicated that corrective actions were sometimes ineffective. Substantial progress has been made in addressing the specific issues.

A total of 88 Licensing Action Requests were issued as part of the GPU Nuclear Licensing Information Tracking System to track identified open items. As of February, 1997, 61 items have been closed.

The overall results of the inspection provided a measure of assurance that the ESW system was capable of performing its intended safety function and was operated, tested and maintained in accordance with its design bases. Therefore, it was concluded that ESW system configuration control and performance was consistent with the design bases and that design basis requirements had been effectively translated into operating, maintenance and testing procedures.

Attachment 4

NRC Vertical Slice Inspections

This attachment provides details on the following NRC Inspections:

1. NRC Safety Systems Outage Modification Inspection (SSOMI)

2. NRC Containment Spray/Emergency Service Water Safety System Functional Inspection (SSFI) - Inspection Report 89-80

3. NRC Electrical Distribution System Functional Inspection (EDSFI) - Inspection Report 92-80

NRC Vertical Slice Inspections

1. NRC Safety Systems Outage Modification Inspection (SSOMI)

A SSOMI was conducted at Oyster Creek in the fall of 1988 involving projects implemented during the Cycle 12R refueling outage. It was conducted in two parts: a design portion and an installation and test portion. As stated in the cover letter accompanying Inspection Report 88-202 dated February 17, 1989, "The purpose of the design portion of the SSOMI was to examine, on a sampling basis, the detailed design and engineering which was required to support modifications implemented during the outage." As stated in the cover letter accompanying Inspection Report 88-203 dated March 16, 1989, "The purpose of the installation and test portion of the SSOMI was (1) to determine, through an examination of specific work packages, that installation of the selected modifications conformed to design and installation requirements and (2) to verify that the repaired or modified components and systems have the required operating configurations and have been adequately tested to ensure that they are capable of safely performing their intended functions".

The stated purpose of each portion of the SSOMI, when combined, address items (b) and (c) of the NRC's information request. The SSOMI examined selected projects from design through installation and testing to determine that design basis and procedural requirements were properly addressed.

SSOMI Design Portion (Inspection Report 88-202)

Twenty eight modifications were reviewed by the inspection team. Of these, no deficiencies were found for eleven of the modifications. Twenty six deficiencies were noted among the other seventeen modifications.

Five deficiencies identified by the inspection team were considered to have sufficient safety significance to justify their classification as items requiring resolution prior to restart from the current refueling outage. GPU Nuclear responded to the restart concerns in letters dated December 12, 1988 and January 19, 1989. Resolution of the deficiencies for restart consisted of performing additional analysis and/or providing additional operator guidance. Resolution of the other deficiencies involved revising calculations, procedures and the FSAR, and, in a few cases, modifications.

In addition to identifying deficiencies, the inspection team "...identified several programmatic strengths which contribute to the effectiveness of your design effort. Safety evaluations were generally comprehensive and well documented, as were the Modification Design Descriptions reviewed during the inspection. The reviews indicated that you have generally been successful in controlling design interfaces, and many modification packages reviewed during the inspection were determined to be free of apparent deficiencies. Your initiatives to

conduct Safety System Functional Inspections and develop a configuration management program are considered positive steps in effective management of the design process." (Inspection Report No. 88-202, cover letter dated February 17, 1989, page 2, paragraph 2).

GPU Nuclear submitted a response to open design portion deficiency issues, as requested, on May 31, 1989. No enforcement actions were taken as a result of the design portion of the SSOMI.

SSOMI Installation and Test Portion (Inspection Report 88-203)

In this portion of the inspection, the team reviewed the implementation of fifteen modifications. In addition, plant material condition was assessed and work control processes were reviewed. Almost all findings during the installation and test portion of the SSOMI were related to work control issues. By letter dated January 12, 1989, GPU Nuclear responded to an NRC letter dated December 23, 1988 which requested a response to four issues. One issue concerned IST procedures which contained isolation valve closure time acceptance criteria less conservative than the design basis described in the FSAR. Measured closure times also exceeded the FSAR values for some of the valves. A second issue also involved the FSAR in that certain control room ventilation flow design values were not met upon testing of a system modification.

As discussed in the inspection report, GPU Nuclear committed to certain actions to improve FSAR accuracy and assess the impact the identified errors may have had on previous plant change activities.

2. NRC Containment Spray/Emergency Service Water Safety System Functional Inspection (SSFI) - Inspection Report 89-80 (August, 1989)

This NRC inspection was conducted by NRC Region I and contractor personnel. The inspection was performed to assess the operational capability of the containment spray (CS) and emergency service water (ESW) systems to perform their design basis safety functions. The inspection was divided into the areas of mechanical, electrical, instruments and controls, operations, maintenance, and surveillance and testing. The inspection also assessed engineering support and involvement, and management support and QA involvement in activities related to the CS/ESW systems.

The inspection team evaluated the adequacy of operational procedures, test practices, and maintenance policies related to CS and ESW reliability, and also addressed the quality of design control and other management programs applied to the CS and ESW systems. This was accomplished by determining:

- the capability of CS and ESW to perform the safety functions required by the design bases,
- the adequacy of testing to demonstrate required safety functions,
- the adequacy of maintenance to ensure reliable system availability,
- the adequacy of human factors considerations related to CS and ESW supporting procedures, and operator training to ensure proper system operation,
- the effectiveness of quality program activities related to CS and ESW in identifying safety issues and in assuring their resolution, and
- the effectiveness of engineering support including calculations in providing assurance that the systems, as operated, meet the design bases.

The inspection team reviewed the FSAR, Technical Specifications, modification packages, and design documentation supporting design and operation of the CS and ESW systems including safety evaluations, calculations, engineering procedures, technical data reports and procurement specifications. CS and ESW system operating procedures and maintenance procedures and programs including IST were evaluated, and control room activities were observed. System configuration and consistency with design documents were verified by system walkdowns. The adequacy of overall design control programs as applied to CS and ESW was evaluated.

The SSFI team reviewed and evaluated the adequacy of engineering in maintaining and updating the design bases, translating the design bases into operating procedures and controlling design changes. Elements of the ISI Program, Engineering Procedure Update Programs, and the commitment to the development of a design basis document program and internally performed SSFI's were identified as specific strengths. System walkdowns confirmed that the general configuration of the systems matched the FSAR description. With the exception of identified findings, the inspection determined that, in general, changes

resulting from modifications, information in modification design descriptions, and FSAR changes were generally found to be reflected in procedures.

Class IE electrical systems were determined to be capable of providing for the acceptable operation of the CS and ESW systems. The instrumentation and control review identified the failure modes and effects analyses for the CS system as a strength in identifying potential single failures in automatic/manual controls and supporting power supply system. Several calculational errors and problems were identified but were not considered to adversely effect system operability. Overall, the inspection determined that the CS and ESW systems are functionally capable of performing their design requirements from an instrumentation and control perspective.

The inspection team verified that the components were properly installed and matched the equipment shown on the drawings and equipment data listings. Maintenance documentation was reviewed and included general maintenance procedures, history records, maintenance scheduling, specific work job orders, discrepancy reports, LER commitments and vendor data. Specific review of the CS/ESW heat exchangers concluded that the maintenance performed provides reasonable assurance that the heat exchangers will perform with adequate efficiency to satisfy functional requirements. Instrument and control maintenance practices reviewed were determined to be satisfactory. Overall the inspection noted that the material condition of the ESW and CS system components was being effectively maintained and that specific predictive maintenance practices were identified as a strength.

CS and ESW operability and IST procedures were evaluated and determined to be adequate for determining the minimum required system flow rate. The inspection determined that the overall system surveillance is adequate to verify system operability with the exception of heat removal capacity heat exchanger testing.

Evaluation of engineering support activities verified that system engineers were knowledgeable of system functions and that significant mechanical systems calculations and studies have been developed for the CS/ESW systems. The inspection team also recognized that the Design Basis Document for the CS system was being developed. However, the inspection identified that engineering support and involvement with plant organizations needs improvement to fully translate the design basis into plant operability procedures and plant IST procedures.

The inspection concluded that the CS and ESW systems are functionally capable of performing their design basis safety functions. The inspection identified specific weaknesses which included lack of a commitment to a calculation update program, instances where operating, surveillance and test procedures did not reflect adequate transfer of design basis information, and instances where inadequate FSAR updates and unclear Technical Specifications caused difficulty in properly performing safety evaluations. Inspection findings in identified performance areas provided valuable insight as to where additional emphasis on design basis controls were needed.

Identified Performance Weaknesses (Violations):

- Lack of calculations to demonstrate ability of CS/ESW to transfer the heat load from containment in the event of a LOCA with ESW flow throttled and with elevated canal temperature.
- FSAR not updated in a timely manner to reflect CS and ESW system design basis alignment, operations, and analyses.
- CS system logic temporary modification - operators were not aware of changes, and change was not properly posted in affected procedure.
- Incomplete implementation of IST surveillance procedure.
- ASME Code, Section XI required alert range limits for ESW pumps deleted or revised without appropriate NRC relief.
- Inadequate measures for identification and control of the heat exchanger relief valves.
- EDG load calculations not consistent with FSAR commitment to AEC Safety Guide 9.
- Single failure susceptibility of CS system control circuits not consistent with FSAR single active failure design bases.

3. NRC Electrical Distribution System Functional Inspection (EDSFI) - Inspection Report 92-80 (May, 1992)

An NRC inspection team conducted an electrical distribution system (EDS) functional inspection to determine if the EDS was capable of performing its intended safety functions as designed, installed and configured. A second objective was the assessment of GPU Nuclear engineering and technical support for EDS activities.

The inspection reviewed calculations, design documents and test data and focused on those attributes which ensure quality power is delivered to those systems and components that are relied upon during and following a design basis event. Procedures and guidelines governing the EDS design calculations, design control, and plant modifications were also reviewed.

This inspection covered portions of on-site and off-site power sources and included the 230KV and 34.5KV off-site power grids, main transformers, unit startup and auxiliary transformers, 4.16KV normal and emergency busses, emergency diesel generators (EDG), 480V safety-related unit substations and MCC, station batteries, battery chargers, inverters, 125VDC safety-related busses, and the 120Vac vital distribution system. The inspection verified the adequacy of the emergency on-site and off-site power sources for the EDS equipment by reviewing regulation of power to essential loads, protection for calculated fault currents, circuit independence, and coordination of protective devices. The adequacy of mechanical systems which interface with and support the EDS was verified. Selected EDS equipment configuration and rating was also verified. Additionally, maintenance, calibration and surveillance activities for selected EDS components were reviewed.

The inspection considered conformance to General Design Criteria (GDC) and other regulatory requirements, and commitments contained in Technical Specifications, the FSAR and appropriate safety evaluation reports. The EDS design was evaluated for compliance with specifications, industry standards, and regulatory requirements and commitments.

The inspection assessed the effectiveness of the controls established to ensure that the design bases for the EDS were being maintained. As-built configuration was verified to be consistent with design documents and modification packages. Inspection attributes for plant modifications included assessment of the design review process and the resulting safety evaluations to meet the requirements of 10 CFR 50.59. The review of selected modifications demonstrated that cable sizing procedures and installation requirements had been properly evaluated. Adequate margin between nameplate rating and analyzed load requirements was confirmed for installed equipment reviewed. A review of connected loads versus calculated loads confirmed the adequacy of the EDS design. Review of equipment specification performance ratings verified consistency between performance requirements and ratings. Plant design changes and modifications were reviewed to verify that changes to the plant are controlled and performed in accordance with approved procedures and in accordance with regulatory requirements. Engineering packages were determined to be thorough, well-documented and appropriately evaluated under 10 CFR 50.59. Temporary modifications

3. NRC Electrical Distribution System Functional Inspection (EDSFI) - Inspection Report 92-80 (May, 1992)

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The inspection reviewed calculations, design documents and test data and focused on those attributes which ensure quality power is delivered to those systems and components that are relied upon during and following a design basis event. Procedures and guidelines governing the EDS design calculations, design control, and plant modifications were also reviewed.

This inspection covered portions of on-site and off-site power sources and included the 230KV and 34.5KV off-site power grids, main transformers, unit startup and auxiliary transformers, 4.16KV normal and emergency busses, emergency diesel generators (EDG), 480V safety-related unit substations and MCC, station batteries, battery chargers, inverters, 125VDC safety-related busses, and the 120Vac vital distribution system. The inspection verified the adequacy of the emergency on-site and off-site power sources for the EDS equipment by reviewing regulation of power to essential loads, protection for calculated fault currents, circuit independence, and coordination of protective devices. The adequacy of mechanical systems which interface with and support the EDS was verified. Selected EDS equipment configuration and rating was also verified. Additionally, maintenance, calibration and surveillance activities for selected EDS components were reviewed.

The inspection considered conformance to General Design Criteria (GDC) and other regulatory requirements, and commitments contained in Technical Specifications, the FSAR and appropriate safety evaluation reports. The EDS design was evaluated for compliance with specifications, industry standards, and regulatory requirements and commitments.

The inspection assessed the effectiveness of the controls established to ensure that the design bases for the EDS were being maintained. As-built configuration was verified to be consistent with design documents and modification packages. Inspection attributes for plant modifications included assessment of the design review process and the resulting safety evaluations to meet the requirements of 10 CFR 50.59. The review of selected modifications demonstrated that cable sizing procedures and installation requirements had been properly evaluated. Adequate margin between nameplate rating and analyzed load requirements was confirmed for installed equipment reviewed. A review of connected loads versus calculated loads confirmed the adequacy of the EDS design. Review of equipment specification performance ratings verified consistency between performance requirements and ratings. Plant design changes and modifications were reviewed to verify that changes to the plant are controlled and performed in accordance with approved procedures and in accordance with regulatory requirements. Engineering packages were determined to be thorough, well-documented and appropriately evaluated under 10 CFR 50.59. Temporary modifications

were also found to be properly evaluated under 10 CFR 50.59. The inspection concluded that GPU Nuclear had implemented adequate measures to effectively control system configuration. These results confirm that the controls established for plant modifications were technically adequate and thus maintained plant design basis requirements.

Although margins were considered small on the 4KV bus load, overall the AC short circuit analyses confirmed the adequacy of equipment design and ratings. Station batteries were determined to have adequate capacity and were adequately sized. Several weaknesses were identified in the knowledge of the design bases of the EDS as demonstrated by omission of ac voltage analyses at the component level for many safety related devices, the absence of a 125 Vdc system voltage drop study, and deficiencies in safety related 4 kv and 460 V switchgear room HVAC alarms and procedural guidance. These areas have been addressed and it was further noted that positive measures had been taken to assure reliable off-site power including periodic grid stability studies. Electrical protection system under-voltage and over-voltage setpoints were determined to be adequate to protect the RPS equipment for under-voltage and over-voltage conditions.

Station procedures and electrical lineups for various bus alignment modes were reviewed and confirmed to be in conformance with the Oyster Creek FSAR.

The inspection determined that the EDGs, including support systems, were adequately designed, well maintained and historically reliable. Maintenance and testing of the EDGs was found to be generally adequate. Programs for the control of maintenance and test equipment were effective. EDG surveillance tests and setpoint calibrations were verified to be adequate. EDG surveillance test procedures were reviewed and confirmed to adequately verify proper sequencing of emergency loads onto the EDG for the design basis accident scenarios. Inspection and testing of circuit breakers, including over-current protective settings, were verified to be adequate. The identified lack of functional testing of station battery components has been addressed. These results provide reasonable assurance that EDS maintenance and testing programs have adequately maintained plant design basis requirements.

Evaluation of self-assessment programs, which involved activities including safety system functional inspections, QA audits and surveillance, determined these programs to be a strength.

The inspection concluded that the electrical distribution systems at Oyster Creek are capable of performing their intended safety functions based on the design documents reviewed and equipment inspected. Further, the overall results of this inspection, in conjunction with resolution of identified weaknesses provide a measure of assurance that plant system, structure, component configuration and performance are consistent with the plant design bases and that design bases requirements are adequately translated into operating, maintenance and testing procedures; and that adequate controls, programs and procedures are implemented in order to effectively maintain the plant design bases requirements.

Attachment 5

Problem Identification and Corrective Action Processes

This attachment provides a detailed description of problem identification and corrective action processes for Oyster Creek.

Problem Identification and Corrective Action Processes

Discussion

GPU Nuclear employs a variety of methods to identify problems, determine their extent, take corrective action to prevent recurrence and make required reports to the NRC. Corporate Policies have the highest level of authority within GPU Nuclear's document system[1]. The following policies outline the basic approach and operating philosophy that apply throughout the company concerning identifying problems and implementing corrective action.

- GPU Nuclear Corporation Quality Assurance Policy, 1000-POL-7200.01
- Quality of Work Policy, 1000-POL-1000.02
- Regulatory Compliance, 1000-POL-1740.03
- Use of the Ombudsman Function, for Resolving Nuclear or Radiation Safety Concerns, 1000-POL-1020.01

The Quality Assurance (QA) policy[2] establishes the intention of GPU Nuclear to comply with the Code of Federal Regulation, the NRC Operating Licenses and the appropriate codes, guides and standards with respect to operation, in-service inspection, maintenance, procurement, repair, and modification of the stations. This policy requires all personnel and contractors to comply with the requirements established in the implementing QA plan[3] and establishes the Director-Nuclear Safety Assessment (NSA) as the individual with the overall authority and freedom to identify problems. This authority includes the authority to stop work and to recommend unit shutdown.

The Quality of Work policy[4] requires every organizational unit to strive to improve the work environment to encourage employees to freely admit mistakes and bring problems out in the open. This policy also stresses the importance of identifying the causes of undesired incidents and implementing effective corrective action in a timely and accountable manner. Similarly, the Regulatory Compliance policy[5] states that employees are responsible for reporting to their supervision conditions or situations that are not in accordance with Company or legal requirements. A corporate policy[6] also describes the Ombudsman function which has been established to promote a high degree of access for any individual with a concern for nuclear or radiation safety. This policy also reiterates that workers have the right and responsibility to identify and report safety concerns.

[1] GPU Nuclear Policy, Plan and Procedure System. 1000-ADM-1218.01, Revision 8
[2] GPU Nuclear Corporation Quality Assurance Policy, 1000-POL-7200.01, Revision 2
[3] GPU Nuclear Operational Quality Assurance Plan, 1000-PLN-7200.01, Revision 8
[4] Quality of Work Policy, 1000-POL-1000.02, Revision 1
[5] Regulatory Compliance, 1000-POL-1740.03, Revision 3
[6] Use of the Ombudsman Function, for Resolving Nuclear or Radiation Safety Concerns, 1000-POL-1020.01, Revision 4

These policies are implemented by plans and procedures. The quality assurance program[7] establishes the following general requirements concerning nonconforming conditions and corrective action: (1) nonconforming materials or activities within the scope of the GPU Nuclear Quality Assurance Program shall be identified and controlled to prevent their inadvertent utilization; (2) measures shall be established which ensure that conditions adverse to quality are promptly identified and corrected; (3) the cause of significant conditions adverse to quality shall be determined and appropriate action taken to prevent recurrence; (4) the identification, cause, and actions taken to correct significant conditions adverse to quality shall be documented and reported to the appropriate levels of management; (5) significant conditions within the intent of 10 CFR 21 shall be reported to appropriate management levels within the affected organization for review and evaluation; and (6) some deficiencies can be promptly corrected without initiating defined deficiency and/or nonconformance reports. Such deficiencies are typically those which are isolated to singular occurrences, not repetitive in nature, and/or are such that appropriate action to prevent recurrence can be initiated at the time the deficiency is identified and do not require any action other than reporting the occurrence. These deficiencies shall be periodically analyzed to detect adverse trends as may be present. The results of analyses shall be periodically reported to management for review and assessment. When significant conditions are identified or when actions are required by upper management to correct problems, such as a generic problem identified by the trend analysis or repetitive failure to disposition nonconformances, these problems shall be elevated to upper levels of management for resolution.

These requirements are implemented by the following procedures which provide various and overlapping opportunities/methodologies to identify problems by line employees, their management, independent oversight (NSA) personnel, external senior review panels (i.e. General Office Review Board and Nuclear Safety Compliance Committee) and through trending and analysis:

- Nuclear and Radiation Safety Plan[8]
- Stop Work Notification[9]
- Shutdown Recommendation/Directive[10]
- GPU Nuclear Corrective Action Programs and Processes[11]
- NRC Regulation 10-CFR-21 (Reporting of Defects and Noncompliance)[12]
- Control of Nonconformances and Corrective Action[13]
- GPU Nuclear Quality Deficiency Reports (QDR)[14]
- Licensee Event Reports (LER)[15]

[7] GPU Nuclear Operational Quality Assurance Plan, 1000-PLN-7200.01, Revision 8
[8] GPU Nuclear Corporate Policy and Procedure Manual, 1000-PLN-1291.01, Revision 6
[9] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7202.01, Revision 2
[10] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7202.02, Revision 2
[11] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7216.01, Revision 1
[12] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-1290.01, Revision 5
[13] Oyster Creek Nuclear Generating Station Procedure, 104, Revision 24
[14] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7215.02, Revision 5
[15] Oyster Creek Nuclear Generating Station Procedure, 106.1, Revision 7

- Receipt Deficiency Reports[16]
- Nuclear Safety Assessment (NSA) Audit Program[17]
- Management Escalation Program for NSA Quality Deficiencies[18]
- Incident Critique Procedure[19]
- NSA Monitoring Program[20]
- Deficiency Trend Analysis[21]
- NSA Vendor Surveillance[22]
- Independent Onsite Safety Review Group Procedure[23]
- Safety Issues Assessment Program[24]
- OQA Plan Document Reviews[25]
- Supplier Corrective Action Request[26]
- Access Authorization and Fitness-For-Duty Program Audits of Contractors[27]
- NSA Vendor Audit Program[28]

Other opportunities also exist for identifying problems. Safety System Functional Inspections have been performed and a program for reconstituting design basis documentation is in progress. These programs have been very effective in identifying questions and problems related to the design basis. Operating experience from international and domestic nuclear plants is continually received and evaluated for applicability to Oyster Creek. Also, GPU Nuclear participates in the Cooperative Management Audit Program (CMAP) which is a 13 member industry organization that provides a biennial audit of the effectiveness of Oyster Creek's quality assurance program including corrective action. The audit team is made up of qualified lead auditors from two or more utilities and the audit duration is typically two weeks. CMAP may also provide technical specialists to participate in GPU Nuclear audits; these technical specialists bring a fresh but experienced eye to review GPU Nuclear activities and provide a better capability for discovering long standing but unrecognized problems. Similarly the Institute for Nuclear Power Operations (INPO) performs periodic reviews to identify problem areas. Also when employees leave the company, an exit interview is conducted and the employee is questioned as to whether they have any nuclear or radiological safety concerns.

When problems are identified they are entered into an approved corrective action system. The system most frequently used at Oyster Creek is the Deviation Report[29], although reporting by

[16] Oyster Creek Nuclear Generating Station Procedure, 125.2.10, Revision 1
[17] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7218.01
[18] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7216.01, Revision 0
[19] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-1201.01, Revision 2
[20] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7210.03, Revision 2
[21] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7201.02, Revision 0
[22] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7207.04, Revision 1
[23] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-1010.01, Revision 3
[24] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-1010.03, Revision 2
[25] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7206.01, Revision 0
[26] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7216.02, Revision 1
[27] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7218.03, Revision 0
[28] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7218.04, Revision 0

another approved and similar system such as LER or QDR is also acceptable. The Deviation Report (DR) system serves to provide reasonable assurance that events and situations which may require further review, reporting, or corrective action are identified, controlled, documented, reported, and evaluated. The DR system also requires that necessary corrective action be determined, assigned, tracked, and completed. The process also includes measures to promptly inform operating supervision of deviations which could affect operability so that operability determinations can be made. A multidiscipline review group is used following the operability determination to review the operability declaration and basis, if made, determine the significance of the issue, the priority/level of the root cause determination, and the need for reporting the issue to industry or the NRC. This group then assigns the action and commences to follow the issue to promote timely close-out. A safety review engineer independently ensures that necessary corrective actions have been determined, assigned, and entered into a tracking system and must concur with the recommendation to close out the issue. For significant deficiencies, the Manager NSA must concur that the root cause determination is complete and corrective action has been implemented prior to closing out the issue. The safety review engineer periodically trends DRs to identify repetitive failures and to initiate corrective action to prevent recurrence. DRs are also evaluated to determine if a material nonconformance exists. If so, a Material Nonconformance Report (MNCR) is issued to ensure the material is identified and controlled to prevent inadvertent use.

A Quality Deficiency Report (QDR) may also be used to document deficient conditions. A QDR can be initiated by any individual identifying a quality deficiency. The QDR is reviewed by the NSA Manager or designee for reportability and an action party is assigned. The action party determines the cause and extent of the deficient conditions, and the actions necessary to prevent recurrence. The NSA Manager reviews and concurs with the root cause and proposed corrective action or requires a revised response by the action party. NSA tracks the corrective action and verifies completion prior to closing out the QDR.

Irrespective of the reporting system (DR, LER, or QDR) for significant problems, the analysis for root cause is determined. More than 70 personnel have received training in techniques such as, the NRC Human Performance Investigation Process, Human Performance Evaluation System, Management Oversight and Risk Tree, Gallo Root Cause Program and Kepner Tregoe's Problem Analysis and Decision Making.

Compliance with the requirements specified in the controlling plans and procedures is periodically assessed and reported to management by certified auditors independent from the activities reviewed. Also providing oversight are external organizations such as audit teams from the Cooperative Management Audit Program and evaluation teams from INPO and the National Academy for Nuclear Training. Also senior industry personnel make up GPU Nuclear's General Office Review Board (GORB) and Nuclear Safety Compliance Committee (NSCC). The GORB reviews and reports annually on the effectiveness of the independent oversight activities of the NSA department to the President, GPU Nuclear. The NSCC also overviews activities and reports their findings to the GPU Nuclear Board.

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[29] Oyster Creek Nuclear Generating Station Procedure, 104, Revision 24

Future Plans

Efforts are underway to enhance the current corrective action processes. The enhancements improve and simplify the problem identification mechanism such that there will be more consistency between Oyster Creek and TMI. The trending process has also been enhanced to provide more useful information concerning human performance.

Attachment 6

Commitments

This attachment provides a list of all commitments being made in this submittal by GPU Nuclear for the Oyster Creek Nuclear Generating Station.

Commitments

This following table provides a list of all commitments being made in this submittal by GPU Nuclear for the Oyster Creek Nuclear Generating Station.

Commitment Number	Description	Completion Date
1	Develop Engineering Division Procedure EP-045 to improve the review and update process in the following areas: • Establish a periodic review of the SDBDs and FSAR to ensure consistency. • Provide improved process for SDBD maintenance and control. • Develop improved guidelines and process for biennial FSAR update.	4/1/97
2	Consolidate resolution tracking for SDBD and SSFI open items by entering them into the Deviation Report process to achieve a timely and effective resolution.	6/30/97
3	Complete procedure revisions for and training on an improved process for control and update of calculations and TDRs.	9/30/97
4	Utilize established processes to evaluate those systems under consideration to establish future design basis reconstitution activities.	9/30/97
5	Conduct personnel training on design bases issues and the new FSAR update process (EP-045).	12/31/97
6	Develop a specific plan and schedule for improving setpoint basis documentation / references.	5/30/97
7	Complete five (one partial) and upgrade two SDBDs.	12/31/97
8	Develop and implement a one time detailed FSAR review.	9/30/98

SDBD = System Design Basis Document
TDR = Technical Data Report

Attachment 7

Oyster Creek Design Bases Reconstitution Program

This attachment provides a detailed description of GPU Nuclear's Oyster Creek Design Bases Reconstitution Program.

Oyster Creek Design Bases Reconstitution Program

I. SUMMARY

The GPU Nuclear Design Bases Reconstitution Program has been effective in improving our knowledge and the availability of our most important design bases information. This has been achieved by locating supporting design information and design process documentation. This has resulted in:

- Providing insights into the fundamental design philosophies and intents.
- Identifying gaps in the design documents, design process documents and supporting design information.
- Identifying changes to operating, maintenance, and testing procedures.

The results derived from conducting Oyster Creek design bases reconstitution program include:

- The core group of the System Design Basis Documents (SDBD) completed provides a comprehensive source of design bases information in a centralized format.
- The GPU Nuclear conducted SSFIs have determined that modifications performed have preserved the design basis of the Core Spray and Automatic Depressurization System.
- The extensive search and review of GE Nuclear Energy and Burns & Roe files, by GPU Nuclear, has identified and located specific document types which provide a substantial portion of the original engineering design bases and associated supporting design information for Oyster Creek Systems.

II. DESCRIPTION - GPU Nuclear DESIGN BASES RECONSTITUTION PROGRAM

A. GPU Nuclear Engineering Management Strategic Perspective

1. GPU Nuclear Configuration Management Program

In 1988, as a result of both industry and regulatory initiatives, the initial version of the GPU Nuclear Configuration Management policy was issued. That policy directly recognized the need for a defined and documented design basis.

The Design Bases Reconstitution Program represented the initiation of a deliberate search for supporting design information and the collation of that information, in the design basis document. This program added the design basis document to GPU Nuclear's core set of engineering documents and the Safety System Functional Inspection technique as a means to confirm that the physical plant and the manner it is operated, maintain and altered conformed to the design basis document.

GPU Nuclear did have design basis information, including some of the supporting design information, in the form of design documents, project correspondence, memoranda, GPU Nuclear personnel knowledge, original designers, etc. The purpose of this program was to locate, selectively procure, collate and selectively reconstitute design basis information to ensure GPU Nuclear had it, knew it, and placed it where it was accessible.

Additional initiatives were undertaken to improve and expand design process documents to also provide or confirm the design basis information. These initiatives included extensive electrical system studies and calculations (e.g. short circuit analyses, degraded grid studies, breaker coordination studies, etc.), instrument loop error calculations, control of setpoints, plant transient/cycle logging, etc. Such initiatives have augmented the design basis documents that have been developed to provide a comprehensive body of design basis information.

2. **GPU Nuclear System Engineering Function**

The system engineer/owner function is intended to provide or coordinate technical support to operations and maintenance on assigned systems. The system engineer/owner is also the owner of the Design Basis Documents and responsible for maintaining assigned SDBDs current.

B. Design Bases Reconstitution Program

1. Objective

The current GPU Nuclear Design Bases Reconstitution Program transitioned from a pilot to a fully defined project in the fourth quarter of 1989. The objective of the program was to locate as much of the engineering design bases and associated supporting design information for the selected systems as possible; and, capture and disseminate that information using the Design Basis Document.

2. Selection of Systems/Structures/Components

The criteria that has been used to select the systems and topics to be included in the design basis reconstitution program and to formulate the schedule for completing the program has changed over time as needed to address GPU Nuclear's needs. Some of the critical criteria utilized were:

- safety significance of the system;
- risk significance of the system based upon probabilistic risk assessment results;
- potential for future modifications for which detailed design bases information was needed;
- retirement of the key original designers of the systems;
- existence of system challenges/difficulties that frequently occurred;
- level of corrective and preventative maintenance that was required.

The other major consideration used was the extent of the need for design bases information to support:

- system engineering functions;
- system performance team activities;
- reliability centered maintenance/life of system maintenance programs;
- plant life extension/license renewal; and,
- conduct of Safety System Functional Inspections or equivalent "vertical slice" system assessments;

3. Role - GPU Nuclear Engineers

From 1989 to 1993, GPU Nuclear's Design Basis Reconstitution Program was highly dependent on the use of contracted engineering and technical support personnel. During this period the need for such contracted support was predicated on the following attributes:

- unique access to design basis documentation (e.g. GE original files);
- direct access to original system designers;
- availability of capable engineers who could be contractually dedicated to search for and understand original engineering design basis information; and,
- availability of capable engineers who could prepare design basis documents.

GPU Nuclear engineering personnel involvement included task and contract management, identification of key issues / problems, coordination of overall scope and technical reviews.

By 1994, the GPU Nuclear Design Bases Reconstitution Program had matured to the point that GPU Nuclear employees are capable of preparing Design Basis Documents.

4. Design Basis Document Content

GPU Nuclear's approach was to provide an up-to-date engineering document in which would reside the engineering design bases information with references to the source documents and important design documents for selected systems.

5. Recovery of Burns & Roe (B&R) Design Information

The recovery of B&R Design Information was conducted in three phases:

- In 1978, B&R was requested to review their files of documents associated with the design and construction of Oyster Creek. The review was conducted to identify those documents which were quality assurance records in the context of ANSI N45.2.9. Once these documents were identified, copies in the form of micro film reels were to be produced. This effort provided a compilation and copy of design documents (design criteria, design specifications, procurement specifications, etc.) and design process documents (calculations, pre-operational tests, material test reports, etc.) This effort did not provide supporting design information.

- Between 1989 and 1992, B&R participated in GPU Nuclear's Design Bases Reconstitution Program as both a researcher and a preparer of SDBDs. B&R reviewed their files and located design process and supporting design information documents. The supporting design information was contained principally in project memoranda.

- In 1995, GPU Nuclear procured the Oyster Creek files from B&R. These files were reviewed by a GPU Nuclear SDBD engineer. This review located additional supporting design information in the form of conference notes, design review comment resolutions, bid evaluations, telecons, etc. Addition design process information, in the form of progress review meeting minutes, audits, etc. was also located.

The end result of this recovery was that a substantial portion of both GE Nuclear Energy and Burns & Roe original design process and supporting design information was recovered

6. Recovery of GE Nuclear Energy (GENE) Design Information

Between 1989 and 1992, GENE participated in GPU Nuclear's Design Bases Reconstitution Program as a preparer of design basis documents. The GPU Nuclear task engineer was accountable to collate the referential basis and provide it to GENE. GENE was to supplement these GPU Nuclear supplied references with documents that were accessible in GENE databases and archives.

In 1995, as part of a study conducted by GPU Nuclear, GENE and Black & Veatch, a detailed search and characterization of GENE documentation was provided by a GPU Nuclear SDBD Engineer. The detailed search focused on locating document types that contained engineering design bases and the associated supporting design information that was not accessible through GENE databases or archives.

GPU Nuclear is currently procuring GENE document types which appear to have the highest value with respect to providing the original Oyster Creek supporting design information. The documents generally consist of:

- original studies, evaluations, analyses, etc. which defined the inputs for transient and accident analyses;
- studies, evaluations, analyses, etc. associated with the sizing of systems and major components;
- engineering change documentation associated with field changes during original construction;
- project correspondence, memoranda, etc. associated with the original design process; etc.

Currently it is anticipated that GPU Nuclear will receive and complete the evaluation of these documents by May 1997.

C. Verification and Validation

1. Verification

The SDBD is a source document for the answer to the question(s) "Why is the system designed the way it is and why is it designed to function as it does?" As such, the SDBD is not required to be design verified as specified by ANSI N45.2.11-1974.

However, the Design Basis Document is intended to provide the current design bases of a system along with the source(s) of those bases. The content of the SDBD is reviewed by GPU Nuclear personnel. The focus of this review is that the SDBD is "complete and accurate within the scope of the document."

2. Validation

Validation, as defined by NUMARC 90-12, is a process that provides reasonable assurance that design bases information is consistently reflected in the physical plant and controlled documents used to support plant operations.

Validation to gain reasonable assurance that design bases information is consistently reflected in controlled documents used to support plant operations is accomplished as part of the current SDBD review process.

The GPU Nuclear Safety System Functional Inspections and Operational Performance Inspection supplemented by results of NRC design reviews and a NRC Electrical Distribution System Functional Inspection (EDSFI) have provided further validation for some SDBDs that design bases information is consistently reflected in the physical plant and controlled documents used to support plant operations.

III. FUTURE DESIGN BASES RECONSTITUTION ACTIVITIES

A. Overview

GPU Nuclear had reevaluated the current status of its Design Bases Reconstitution Program in the Spring/Summer of 1996. The purpose of this reevaluation was to determine if there should be any changes in the systems originally selected or in the schedule (i.e., sequence) of completing the program that had been established. Enclosure 1 lists the design basis documents that have been completed.

Enclosure 3 lists those systems from the original scope and those systems not in the original scope that were being proposed. Each of those systems were evaluated based on the following priorities.

- System Performance Team Priority (SPT)
- Power Uprate Priority (PU)
- License Extension Priority (LE)
- Future Self Assessment Priority (FSA)
- Critical Project Support Priority (CPS)

Further, each of these priorities was weighted to preserve the importance of the needs of the System Performance Team as the primary criteria for selecting a system/topic and determining the schedule for completion.

B. Future Activities

We are continuing to utilize a process involving multi-discipline reviews and prioritization criteria to continuously assess and define the future scope and output of our design bases reconstitution program. The systems listed in Enclosure 3 as "under consideration" are those that we currently feel require additional evaluation to determine if we need to complete a design bases reconstitution. Some factors that we are considering incorporating into our methodology include:

- the extent to which supporting design information is available in vendor manuals and other vendor information;
- the extent to which a topical design bases document can be used to cover a key set of parameters for several interrelated systems (e.g. Cooling Water Systems);
- the extent to which design process documentation (e.g. studies, calculations, etc.) can be used to provide the means to disseminate the design bases and associated supporting design information (e.g. electrical systems); and,
- the extent to which engineering design bases and associated supporting design information already exists in system design descriptions, modification design descriptions, etc,.

The design bases reconstitution effort currently scheduled to be completed in 1997 is provided by Enclosure 2. Enclosure 3 provides an overview of those systems still being considered and those for which design bases reconstitution will not be completed. If the system was risk significant, in the context of the Maintenance Rule, it is being considered even if its overall rank was low.

ENCLOSURE 1

Oyster Creek System Design Basis Reconstitution Activities - Completed

A. Recovery of GE Nuclear Energy (GENE) Design Bases Information

Design bases information associated with the design, licensing, construction and preoperational/startup testing of Oyster Creek, has been located, evaluated and is being selectively procured. The information was located by a extensive search by GPU Nuclear.

It is currently anticipated that original supporting design information of Oyster Creek will soon be accessible. The receipt and evaluation of this information is currently in progress.

B. Recovery of Burns & Roe Inc. (B&R) Design Bases Information

GPU Nuclear has procured design bases information associated with the design construction of Oyster Creek Systems within B&R's scope of supply.

The information has been evaluated for its design bases information content and partially reorganized to improve its retrieveability.

C. Design Basis Documents

Sixteen (16) Design Basis Documents covering twenty (20) systems and one (1) topic have been completed and released for use. Those Design Basis Documents are:

1. SDBD-OC-020 (Partial); "Environmental Qualification"; Draft A.
2. SDBD-OC-212; "Automatic Depressurization System"; Revision 0 (12/22/89).
3. SDBD-OC-212A; "Low Pressure Core Spray"; Revision 0 ( ); Revision 1 (5/3/94).
4. SDBD-OC-220; "Reactor Coolant System"; Revision 0 (4/5/91).
5. SDBD-OC-225; "Control Rod Drive System"; Revision 0 (11/9/94).
6. SDBD-OC-241; "Containment Spray System"; Revision 0 (10/18/89); Revision 1 (8/12/93); Revision 2 (2/6/95); Revision 3 (10/18/95).

7. SDBD-OC-243; "Containment System - Part II"; Revision 0 (6/6/95).

8. SDBD-OC-421; "Condensate and Feedwater Systems"; Revision 0 (9/14/92).

9. SDBD-OC-532; "Emergency Service Water System"; Revision 0 (1/30/91); Revision 1 (10/28/93); Revision 2 (9/26/92); Revision 3 (12/12/95).

10. SDBD-OC-535; "Circulating Water System, Vacuum Priming System"; Revision 0 (10/31/91).

11. SDBD-OC-641; "Reactor Protection System; Revision 0 (1/8/92)"; Revision 1 (9/28/92); Revision 2 (7/18/95).

12. SDBD-OC-661; "Radiation Monitoring System"; Revision 0 ( ); Revision 1 (10/31/94).

13. SDBD-OC-740; "Emergency Power System"; Revision 0 (1/28/91).

14. SDBD-OC-822; "Standby Gas Treatment System, Secondary Containment"; Revision 0 (12/20/91), Revision 1 (9/7/93), Revision 2 (10/12/94).

15. SDBD-OC-838; "Drywell Cooling System, Drywell Chilled Water System; Drywell Temperature Detection System"; Revision 0 (12/29/92), Revision 1 (6/10/93).

16. SDBD-OC-852; "Plant Compressed Air System"; Revision 0 (11/29/93).

D. GPU Nuclear Safety System Functional Inspections

1. TDR-986; "Emergency Electric Power Distribution System", Revision 0 (8/28/89).

2. TDR-1002; "Automatic Depressurization System", Revision 0 (5/15/90).

3. TDR-1028; "Core Spray System", Revision 0 (3/4/92); Revision 1 (12/15/92); Revision 2 (6/2/93).

4. TDR-1118; "Reactor Protection System", Revision 0 (9/23/93).

5. TDR-1176; "Emergency Service Water System", Revision 0 (1/29/96).

ENCLOSURE 2

Oyster Creek Design Basis Reconstitution Activities To Be Completed

A. Completion of the Processing of GENE supporting design information

Subsequent to receipt, the GENE supporting design information will be reviewed, cataloged, and released for use.

B. Completion of the Cataloging of B&R supporting design information

The B&R supporting design information has been inputted into the GPU Nuclear records management system. However, additional cataloging effort will be completed to make the contents of the file boxes retrievable.

C. Completion of Additional design bases documents

1. Complete SDBD-OC-211; "Isolation Condenser System"
2. Complete SDBD-OC-243; "Containment - Functional Portion";
3. Complete SDBD-OC-331; "Steam Jet Air Ejectors";
4. Complete; SDBD-OC-410 (Partial); "Steam Systems";
5. Complete SDBD-OC-735; "125V Station DC System";
6. Update SDBD-OC-212; "Automatic Depressurization System"
7. Update SDBD-OC-212A; "Low Pressure Core Spray System"

ENCLOSURE 3

Oyster Creek System Design Basis Documents--Under Consideration

DESIGN BASIS RECONSTITUTION SYSTEM PRIORITY - OYSTER CREEK							
System Name	SPT Rank	PU Rank	LE Rank	FSA Rank	CPS Rank	Overall Rank	Action
Fire Protection (RSS)	1	4	2	1	1	1.40	UC
Fuel Pool Storage and Cooling (RSS)	1	4	2	1	1	1.40	UC
Intake Structure/Screenwash/Trash Racks	1	4	2	1	1	1.40	UC
Reactor Vessel and Internals	1	1	2	4	1	1.55	UC
Heater Drains, Vent, and Press. Rel. Monitoring	1	1	4	4	1	1.75	UC
Nuclear Instrumentation	1	2	2	4	4	2.10	UC
Control Room and Cable Room HVAC	1	4	2	4	4	2.30	UC
Electrical Distribution, Vital AC (RSS)	1	4	2	4	4	2.30	UC
Dry Fuel Storage	2	4	2	1	4	2.35	UC
Shutdown Cooling (RSS)	2	4	2	1	4	2.35	UC
Instrumentation, Reactor Plant/Vessel	2	4	2	2	4	2.50	UC
Reactor Bldg. Structure	1	4	4	4	4	2.50	UC
Turbine Bldg. Structure	1	4	4	4	4	2.50	UC
Diesel Generator (Mechanical/Controls)(RSS)	2	4	2	4	4	2.80	UC
Fuel Handling (RSS)	2	4	4	4	4	3.00	UC
Reactor Water Cleanup	2	4	4	4	4	3.00	NTBD
Switchyard/Transformers/4160V (RSS)	2	4	4	4	4	3.00	UC
Reactor Bldg. Closed Cooling Water (RSS)	3	4	2	4	4	3.30	UC
Service Water (RSS)	3	4	2	4	4	3.30	UC
Radwaste Systems	3	4	4	4	4	3.50	NTBD
Reactor and Turbine Bldg. Ventilation	3	4	4	4	4	3.50	NTBD
Reactor Manual Control	3	4	4	4	4	3.50	NTBD
Augmented Off Gas	4	1	4	4	4	3.70	NTBD
Generator and Auxiliaries	4	1	4	4	4	3.70	NTBD
Turbine and Auxiliaries	4	1	4	4	4	3.70	NTBD
Turbine Bldg. Closed Cooling Water (RSS)	4	4	2	4	4	3.80	UC
Post Accident Sampling	4	3	4	4	4	3.90	NTBD
Hydrogen Injection	4	4	4	4	4	4.00	NTBD
Reactor and Turbine Bldg. Shielding	4	4	4	4	4	4.00	NTBD

	SPT	PU	LE	FSA	CPS
Weighting Factors (%)	50	10	10	15	15

LEGEND: (a) SPT = System Performance Team, (b) PU = Power Uprate/Dynamic Model Building related, (c) LE = License Extension Support, (d) FSA = Future Self Assessment Support, (e) CPS = Critical Project Support; (f) RSS = Risk Significant System (Maintenance Rule Related); (g) UC = Design Bases Reconstitution is Under Consideration; (h) NTBD = Design Bases Reconstitution is Not To Be Done.