GPU NUCLEAR

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February 10, 1997
6710-97-2023

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

Dear Sir:

Subject: Three Mile Island Nuclear Station Unit 1 (TMI-1)
Docket No. 50-289
Facility License No. DPR-50
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 Three Mile Island Nuclear Station Unit 1 (TMI-1) 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 that our programs and processes are sufficiently effective to provide reasonable assurance that TMI-1 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.

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If you should have any questions regarding this response, please contact Mr. Paul F. Czaya of our Regulatory Affairs Department at (201) 316-7975.

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, TMI-1
TMI-1 NRC Project Manager

Enclosure

Three Mile Island Nuclear Station Unit 1 (TMI-1)

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 Three Mile Island Unit 1 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.

Three Mile Island Nuclear Station Unit 1 (TMI-1) was issued a facility license on April 19, 1974. After the TMI-2 accident in 1979, TMI-1 underwent an extensive review and public hearing process on design and management issues prior to restart in 1985.

Over the approximately 22 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 as has been approved by the NRC. The OQA Plan is written to meet the requirements of 10 CFR 50 Appendix B as 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 Onsite 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 Documents 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 so that design bases 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.

Finally, the plant review group (PRG) at TMI-1 is another tool that is relied upon to address design bases issues related to operability. This process has been effective in requiring that configuration questions, when identified as needing review by the PRG, are thoroughly examined against the design bases.

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 TMI-1. 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 in the past, 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

TMI-1 has a well developed system for preparing new operating, maintenance and test 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. 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 expanded our knowledge of design bases information which has been reflected in operating, maintenance, and testing procedures. Examples include:

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

2. GPU Nuclear SSFIs and Self Assessment

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. Safety System Functional Inspections (SSFI) and a Service Water System Operational Performance Inspection (SWSOPI) 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/Program Review

The NRC has also conducted vertical slice and program review inspections at TMI-1. These have also indicated that the systems reviewed met the design bases. Design control programs were judged adequate. These observations confirm 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 bases requirements are incorporated into TMI-1 operating, maintenance and testing procedures. The Data Analysis Team at TMI-1 was composed of senior engineering and quality assurance personnel familiar with TMI-1. The team focused their attention on the review of reports and assessments from the most recent years. The team reviewed approximately 100 documents. The list of document types is found in Attachment 2. About 500 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 Makeup and Purification (MU&P) and Emergency Feedwater (EFW). The results showed that design bases 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 observations about how well design bases requirements are translated into operating, maintenance and testing procedures at TMI-1.

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

- Reviewers looking deeply in specific areas (SDBD, SWSOPI, SSFI, NRC Inspection Reports, etc.) have generally found that the design bases are adequately included in procedures.

- The results of the review of FSAR and Technical Specification requirements related to the MU&P and EFW 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). The FSAR is readily available to them on the computer with full word search capability. Reviewers are requalified biennially.

- Internal and external assessments have identified the implementation of the Preventative and Corrective Maintenance Programs to be a strength.

- The FSAR update process needs improvement in the area of assuring consistency between design bases and FSAR commitments.

- Programmatic problems have previously been identified in the implementation of the motor operated valve program.

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 TMI-1. 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 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 and program review inspections which have also identified weaknesses but 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. Design Control Processes

TMI-1 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 TMI-1 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. 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. NRC Vertical Slice Program Review/Inspections

The NRC has also conducted vertical slice and program review inspections at TMI-1. These have also indicated that the systems reviewed met their design bases. Design control programs were judged adequate. These observations confirm 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. 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 TMI-1.

- Reviewers looking deeply in specific areas (SDBD, SSFI, SWSOPI, etc.) have generally found that SSC configuration and performance are consistent with the design bases.

- Modifications to the plant have adequately preserved the design bases.

- Analysis of and action by the engineering organization related to selected high focus problems and modifications are considered a strength. Examples are:

  - Crud patterns on fuel
  - Digital Turbine Control Modification

- Identification of reportable events and resolution of corrective action is prompt and effective.

- Internal and external assessments have identified the Design Basis Document and Inservice Inspection Programs as strengths in our overall configuration management program.

- Additional systems need to be considered and open items need to be resolved in a more timely manner in the Design Basis Document Program.

- Various individual Configuration Control issues have previously been identified. Examples identified by the Data Analysis Team are:

  - Circuit breakers in the racked-out position are not seismically qualified (open industry issue).
  - A reactor coolant pump drain line design calculation and corrective action were deficient (closed).
  - Releases from the Turbine Building sump were based on a non-conservative calculation (closed).
  - Insufficient electrical separation between safety and non-safety circuits existed in a few cases (closed).

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 TMI-1, significant problems are required to be identified and entered into an approved corrective action system such as the Quality Deficiency Report (QDR) or Licensee Event Report (LER). A QDR can be initiated by any individual identifying a quality deficiency. The QDR is reviewed by the Nuclear Safety Assessment (NSA) Manager, or designee, for reportability and an action party is assigned. When the need to report to the NRC is identified, GPU Nuclear utilizes its notification and Licensee Event Report procedures to meet the reporting requirements of 10 CFR 50.72 and 10 CFR 50.73. 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.

For lower threshold issues the Event Capture and Reporting process is used. This process serves to reasonably assure that events and situations which may require further review, reporting, or corrective action are identified, controlled, documented, reported, and evaluated. A senior management review group reviews the event and assigns the issue to a responsible department. The responsible department determines the event category based on consequences or potential consequences of the event, evaluates whether an approved corrective action system is applicable and assigns an action party.

Corrective action programs are 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 for installed or warehouse items are

controlled and tracked using a Material Nonconformance Report (MNCR) to provide reasonable assurance of resolution.

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 the GPU Nuclear General Office Review Board which reviews safety related activities at GPU Nuclear, including corrective action.

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 TMI-1 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 self 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 Request - Items (a) through (d). Further supporting information is summarized in the following paragraphs:

Processes
Responses to NRC Request - Items (a) and (d) describe TMI-1 processes to maintain the plant configuration and design bases. These processes include overall

configuration management, configuration change, safety review and other processes. TMI-1 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 TMI-1 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 to 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 the 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 TMI-1 FSAR is currently in progress to examine parameters and statements and verify that they are adequately translated from design bases documents.

- Conducting training for GPU Nuclear personnel on design bases issues and FSAR update **(Attachment 6, Commitment 4)**.

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 21 design bases documents covering 35 systems, procurement of design information from Babcock and Wilcox and Gilbert-Commonwealth/Parsons, and an ongoing program to evaluate and gather design bases information as needed.

Activities to Verify TMI-1 Configuration Management
Over the years multiple activities have been conducted to verify the configuration of TMI-1 and its adherence to design bases. Some of these are:

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

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 TMI-1. Some of these are discussed in response to NRC Request - Items (b) and (c), including major reviews such as NRC Integrated Plant Assessment Team Inspections, SWSOPI, and NRC Electrical Distribution System Functional Inspection (EDSFI). In general, these reviews have indicated that TMI-1 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 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 that, overall, there is reasonable assurance that the design basis has been translated into operating, maintenance and testing procedures and system, structure and component configuration and performance is consistent with the 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 issues. Some observations in the corrective action area included (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-1
The recently completed design inspection at TMI-1 of the Make-Up and Decay

Heat Systems supports the conclusions reached by other inspections. It was concluded that both Makeup and Purification and Decay Heat Systems meet design and licensing intent. Process changes resulting from this inspection are improvement of calculation and Technical Data Report control and setpoint basis and reference documentation **(Attachment 6, Commitments 3 & 5).**

Summary

GPU Nuclear concludes that the programs and processes in place are generally effective in maintaining the TMI-1 configuration consistent with its design bases. TMI-1 was designed in the 1960s and placed in commercial operation in 1974. For that reason, its engineering design bases are not as readily identifiable as for later vintage nuclear units. Nevertheless, the necessary design bases are 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 1974 and as demonstrated by the results of internal and external assessments, GPU Nuclear has concluded there is reasonable assurance that the TMI-1 design bases are understood, available to those who must access them, and generally reflected in plant documentation. Further, this same experience indicated 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 Basis Document Program

GPU Nuclear has been conducting a Design Bases Reconstitution Program for the last eight years. The program has resulted in the development of 21 Design Basis Documents and the performance of five self assessments to confirm that the physical plant in its operation and maintenance conform to design bases (Attachment 7, Enclosure 1). Four of the Design Basis Documents were prepared by GPU Nuclear internally and others were prepared for GPU Nuclear by the NSSS Supplier (Babcock & Wilcox) or Architect Engineer (Gilbert/Parsons). The GPU Nuclear Design Basis Documents focus on answering the question "Why" functional or parametric requirements for a given system were selected. The program also includes the identification and selected procurement of supporting design information from both Babcock & Wilcox and Gilbert/Parsons.

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

1. insights into the 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 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 maintenance, existence of system challenges, support for system engineers and system performance teams.

- GPU Nuclear created its own specification for the content and approval of design bases documents. The focus is to determine the basis 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 bases documents and improved knowledge and accessibility to original files.

- The Design Bases 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 process.

The results achieved to date include:

- Original engineering design bases retained by Babcock & Wilcox has been reproduced and shipped to GPU Nuclear with a comprehensive database constructed to conduct directed document searches.

- Original engineering design bases and associated supporting design information continues to be retained by Gilbert/Parsons in a form which is accessible.

- Twenty one design basis documents covering thirty five systems have been issued for use. Four Safety System Functional Inspections and one Operational Performance Inspection were conducted.

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 bases document is under consideration (Attachment 7, Enclosure 3). GPU Nuclear plans to continue to assess and define the future scope and output of Design Bases Reconstitution Program activities.

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 organizatic n. This organization is composed of the following eight groups with the follow ing roles in the organization.

- Components and Programs
- Nuclear Fuels
- License Renewal
- Safety and Risk Analysis
- Decontamination and Decc:nmissioning
- 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 SSCs, safety reviews and responsibilities.

2.2 Design Control

Section ⁝ 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 Department. 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, 10 CFR 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 documentation consistent with licensed plant configuration and 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 Processes 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 program 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. 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 TMI-1.

The Safety Determination and 50.59 Review screening document 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 Review (RTR) 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 adversely impact 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 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. After the RTR review, it 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 Responsible Technical Reviewers and Independent Safety Reviewers 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 a given 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 basis for the system, and identifies the detailed requirements (including licensing and regulatory requirements) the system must meet along with the technical basis 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 basis 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, any 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 system design bases which has been extracted from original design documents, previous modification documents, engineering and licensing correspondence. (Refer to Attachment 7)

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 SDDs or MDDs, 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 SDBDs, 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 evaluations. The fuel vendor has performed the safety analysis for all reloads to date. GPU Nuclear is in the process of establishing the capability to perform the safety analysis for future reloads. The GPU Nuclear reload methodologies employed have been approved or are in the process of being reviewed for application to TMI-1 by the NRC. Configuration control processes are employed to provide consistency between the core configuration and performance and the design bases. These processes involve four aspects of core management: reload methodology, plant modifications, core loading and core monitoring.

Description:
The first aspect of configuration control for the core is 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 conforms to approved methodology which provides 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 bases. If results fail to meet licensing criteria, then the reload must either be altered or the necessary Technical Specification changes obtained. The analyses are documented by a calculation report (procedure 5000-ADM-7311.01 and independently design verified (procedure 5000-ADM-7311.02). The computer codes used in these analyses are maintained under configuration control by procedure 5000-ADM-7340.01 to provide consistency with approved methodology. Revisions to the procedures and computer codes are controlled under the GPU Nuclear OQA Plan. Fuel vendors are required to notify GPU Nuclear of changes in their methods and 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 consumables (fuel assemblies, 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. 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 provides consistency between the actual core loading configuration and the design loading. This activity provides consistency between the analyzed core loading and the "as loaded" core configuration (Procedures 1505-1, and 1507-12).

The final aspect of configuration control is core monitoring. Shift Engineering is responsible for fulfilling reactor engineering requirements. These responsibilities include reactivity management, power maneuvering support and analysis, and startup physics testing. Plant procedures control these activities for compliance with the applicable Technical Specifications and operating requirements provided in the cycle specific reports, such as the Reload Report, Core Operating Limits Report (COLR), and Physics Data Manual (PDM). 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 Changes

Scope:
The control and changing of setpoints and the basis for the setpoints are addressed by TMI Administrative procedure PEP-9, 'GMS-2 Setpoint Screen'.

Description:
Setpoints are maintained by individual programs and procedures such as operating procedures, surveillance procedures, drawings, and technical manuals. A setpoint database within the work management database (GMS2) provides accumulation of information pertinent to defining the elements and variables of setpoints.

For nuclear safety related 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 consistent with the GPU Nuclear Safety Review Process.

Support calculations are developed, as required, using procedure 5000-ADM-7311.01, 'Calculations' (EP-006) and a design verification is performed using 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. The Safety Review Process provides an RTR and if applicable, an ISR.

5.6 Equipment Control (Switching and Tagging)

Scope:
The tagout and restoration of equipment is a program that applies to all personnel performing maintenance, inspection, troubleshooting, modification or testing activities requiring electrical or mechanical equipment isolation and control to maintain personnel safety and prevent equipment damage. The program is written to

the requirements of the GPU Nuclear OQA Plan . The TMI implementing procedure is AP 1002, 'Rules for the Protection of Employees Working on Electrical and Mechanical Apparatus.'

Description:
A switching and tagging program is implemented by a reviewed and approved procedure. Switching and tagging is designed to properly control equipment removal from and restoration to service. Tagouts are conducted to protect personnel and plant equipment from potential hazards during maintenance and modification activities. The implementation of a switching and tagging program is designed to achieve a high degree of equipment and personnel safety. The protection of Technical Specification required equipment is provided operability verification of redundant equipment prior to removal from service. The personnel involved in the process meet job specific training requirements. The restoration of critical equipment is independently verified per administrative procedure AP 1067, 'Independent Verification Program.'

Level of Review:
The process is initiated with a request to remove equipment from service. This request is reviewed by the cognizant department foreman to ensure adequate personnel/equipment protection. The initiated application is presented to the licensed SRO for preliminary review. The application is officially produced by the switching and tagging operator. This licensed RO reverifies adequate personnel/equipment protection. The duty Shift Foreman (licensed SRO) reviews the application and confirms personnel and equipment protection, Technical Specifications are not violated by the tagout, and redundant equipment is operable. The equipment is then removed from service and tagged out.

Restoration of components to service receives Operations review. The application is verified to be ready for tag removal. The duty Shift Supervisor/Shift Foreman is responsible to provide the restoration details to the switching and tagging operator to prepare the removal order. The tag removal order is independently verified if critical components are affected. The execution of the order is conducted and the execution independently verified if critical components are affected. The entire restoration is not limited to the boundary valves or isolation devices. The order is written to assure all components on or within the work boundary are properly returned to service.

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 TMI 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. 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 bases 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 ' ` technical and quality requirements. Their inclusion of generic design specifications and part numbers data augment configuration control.

Receipt inspection activities provide checks of purchased material that are 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 are 10 CFR Part 50 Appendix B and ANSI/ASME N45.2.2 requirements.

5.8 Project Management and Approval Process

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 PAMP 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 PAMP 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 achieved by 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, 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 TMI-1. 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 (where applicable) for configuration change is listed as follows 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 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 modified 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 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, but 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: for TMI - Technical Evaluation of Replacement Items (PEP-1C) and Engineering Evaluations (PEP-3). 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.

- Safety Determination/Evaluations (10 CFR 50.59) review requirements - If the alternate replacement could affect the FSAR or safety function, then a safety evaluation / determination shall be performed using the Nuclear Safety / Environmental Impact Determination and Evaluation procedure 5000-ADM-1291.01 (EP-016);

- 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 Review:
Alternate replacements are reviewed and approved by the cognizant engineer and the manager, configuration maintenance and manager, system engineering. Quality Verification review and approval are required for alternate replacements, RR and NSR components and where welding is required.

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 a site specific procedure, - Temporary Modifications and Bypass of Safety Functions (AP 1013).

Safety Review:
This TMI-1 procedure contains 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 procedure.

TMI-1 does not use a screening process and requires that the questions set forth in 10 CFR 50.59 are addressed for all Temporary Modifications. An RTR, who is SRO qualified, must review and approve the temporary modification prior to implementation. A qualified Independent Safety Review is required of the safety evaluation, but the implementation of the temporary modification can proceed with the ISR as a follow-up action.

Design Review:
TMI-1 procedures require that a technical/design review be performed and documented by the Engineering Division. The reviews address technical adequacy of the temporary modification similar to engineering configuration attributes listed in the Configuration Change Procedure (EMP-002) which is used for 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 check that the basis, need and plant condition are still applicable to the temporary modification installation and the documentation that supports it.

Level of Review:
Each temporary modification must be reviewed and approved, at a minimum, by an SRO, a qualified RTR, a cognizant engineer and an ISR.

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 Document 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 Review:
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 FSAR for TMI-1 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 biennially 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 TMI-1 FSAR is Update 13, which was submitted to NRC by letter dated April 15, 1996.

5.15 Plant Procedure Development and Change

Scope:
Development of new procedures or changes to existing procedures that are site specific and used to operate and maintain TMI-1, are addressed in station procedure AP 1001A, 'Procedure Review and Approval', procedure AP 1001B, 'Procedure Distribution Control', and AP 1001D, 'Procedure Preparation'. These site specific procedures implement the requirements contained in 1000-ADM-1291.01 and 1000-ADM-1218.01.

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, using the 'Biennial Procedure Review' procedure (AP 1001K) and are performed 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 reviewed 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 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

TMI-1 Data Analysis Team
Documents Reviewed

TMI-1 Data Analysis Team Documents Reviewed

The following table lists documents reviewed by the TMI-1 Data Analysis Team. Data from these documents was evaluated by the team to assess design bases conformance, and, configuration control and corrective action effectiveness.

Table

	Document	Comment
1.	Plant FSAR Updates (PFU)	Reviewed PFUs for submittal only
2.	QA Annual Assessment	1996, 1995, 1994
	Nuclear Safety Assessment Audits	
	Quality Deficiency Reports	
	Capture Forms	
3.	Regulatory Documents	
	Inspection Report	1996 & 1995 (Sample)
4.	Reportability Documents	
	Licensee Event Reports	1996 & 1995 Sample 1991 to 1994
	Plant Review Group Meetings	Search of Design Bases and FSAR references from 1992 to 1996
5.	Operations Analysis Reviews	1990 to 1993
6.	Design Bases Conformance	
	Design Basis Document Open Items	Sample All
	Safety System Functional Inspections	Internal/External
7.	GPU Nuclear Review Committees	
	Nuclear Safety Compliance Committee	1996 & 1995
	General Office Review Board Open Items	1996 & 1995

Attachment 3

GPU Nuclear SSFIs and Self Assessment

This attachment provides details on the following SSFIs and Self Assessment:

1. TMI-1 Emergency Electrical Power Distribution SSFI
2. TMI-1 Liquid Radwaste SSFI
3. TMI-1 High Pressure Injection System SSFI
4. TMI-1 Low Pressure Injection System SSFI
5. TMI-1 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 (TDR) 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 in 1989 prior to any significant design bases reconstitution effort being completed as described in Attachment 7.

A status of resolution of all open items has been inserted.

1. TMI - Unit No. 1 Emergency Electrical Power Distribution SSFI - July 26 to August 25, 1989

The overall objective was to determine if the system has been operated, maintained, and tested in a manner that would lead to the conclusion that the system would perform as designed.

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 Stations." The overall method for the inspection was to:

- research design bases information for the system;
- research work methods and practices which impact design bases (e.g., modification, operations, maintenance, testing, training, and procurement); and,
- review and assess the design bases, processes, procedures, and finished products potentially affecting implementation of the design bases, plant material condition, knowledge of the design bases by personnel and work in progress that was witnessed by the inspection team.

This approach provided the overall structure for collecting, reviewing, and assessing information.

The GPU Nuclear Specification structured the objective of the SSFI into seven underlying questions that address system functional capability. The overall conclusions, structured around the seven questions, were as follows:

1. The design bases for the system are documented with adequate clarity. Assembly of the design bases for a system or issue usually requires multiple documents, some located in vendor files. The team was always able to obtain the appropriate documents on request, although some difficulty was experienced in obtaining the latest revision of several drawings. The team found some drawing errors, but they were minor with respect to system safety function.

2. The system is capable of performing the functions required by its design bases. Improvements that would increase confidence in system performance were identified. The findings of greatest concern are the tap change implementation on the auxiliary transformers, the diesel engine fires, and the 4.16 KV breaker double charging and inadvertent closure because they are based on operational occurrences at TMI.

3. The modifications to the plant have been well documented and have preserved the design bases for the system. The design bases have been successfully revalidated and substantiated by major backfit issues such as 10 CFR 50, Appendix R (Fire Protection) and 10 CFR 50.49 (Environmental Qualification). Only two instances of incompletely documented or undocumented modifications (the 15 kW heater in the Intake, Screen, and Pump House and the space heater in B diesel generator room) were found and their impact on the design bases for the system is minimal.

4. The operating procedures adequately explain how the system is to be operated to conform to the intent of the design. The procedures are backed up by a training program that is a strength. The overall result is that the conduct of operations is regarded as a strength.

5. Testing adequately demonstrates that the system would perform all of the functions required. The surveillance program adequately verifies system operability. The post-work testing program, however, needs to be more soundly based upon functional criteria and is the largest concern in this area.

6. Maintenance is adequate to ensure operability under postulated conditions. The strengths of the program are the maintenance program document, maintenance planning, and maintenance training. However, programmatic improvements to GMS-2 and post-work testing should be quickly completed. Additionally, people working on systems who impact how the design bases are implemented should better understand the role of the updated FSAR as a governing document.

7. Operator training is a strength in ensuring proper operation of the system. Modifications are incorporated into training in a timely fashion to maintain operators current with plant design.

8. The licensing bases and subsequent revisions are included in system documents. As with the design bases, multiple documents from various sources must be assembled for the bases to be complete.

Nineteen open items were reported for further evaluation. These open items were:

- Evaluate the risk represented by the present configuration to EDG availability. The evaluation should consider NRC IE Notice 84-69 and the TMI-1 licensing position on the "blackout rule." The largest benefit may stem from a change to Technical Specifications to not parallel the remaining EDG during daily testing for the 7 days following the other EDG being declared inoperable.

- Evaluate the appropriate level of control for Class 1E cross-ties and consistently apply the controls to all similar configurations within the Class 1E portion of the system.

- Evaluate the potential benefits of Fairbanks Morse recommendations on prelubrication in the context of their potential impact on the probability and duration of diesel fires. "B" Emergency diesel generator experienced a fire during a surveillance test run during the SSFI inspection on 8/14/89. The major concern is the potential to affect the engine's governor and disable the engine.

- Evaluate the need to change auxiliary transformer taps in the context of current postulated grid performance. The evaluation should thoroughly address the operational disturbances created by the existing high voltage conditions, equipment degradation due to high voltage, the risk of error in the tap change evolution and the risk of loss of offsite power during the time spent with a single offsite power source during the evolution as well as degraded grid considerations.

- Complete and document a disciplined root cause analysis for the 4.16KV breaker failures. The analysis should address potential causes for all anomalies noted with the breakers in order to address the possibility of multiple causal failures. The double charging and inadvertent closure of the 500 HP 350 breakers has remained unresolved since November 1987.

- Evaluate the applicability of IEEE 450 to the voltage deviations experienced during surveillances.

- Perform a DC voltage drop calculation. There is a lack of analysis for DC wire lengths and sizes associated with voltage drop to Class 1E loads.

- Perform a 10 CFR 50.59 evaluation for the 15KW heater. The evaluation should address the basis for the temperature specification for the intake screen, and pump house in Chapter 9 of the FSAR and any other design basis impacts on systems, structures and

components in the area (e.g., seismic considerations for a Class II component near a Class I component).

- Change the procedures to stop the practice of testing solvents on nuclear safety related components.

- Evaluate the applicability of generic or specific toggle specs for the use in PM procedures and utilize the results in the procedures.

- Evaluate the benefits of recalibrating relays and other components that are acceptable in the as found condition and near their plus or minus limits.

- Modify procedures to ensure specifications of even small parts needed for modifications.

- An evaluation should be performed to assess the types of data that should be part of a trending program.

- Post work testing program should provide guidance on assessing active and passive functions and determine what functions have been impacted by work and need to be restated.

- Training for licensing basis documents should be evaluated to assure those who impact design and legal requirements understand those requirements.

- Evaluate design document control mechanisms and facilities from the perspective of those who rely on up-to-date design information.

- During follow-up on the evaluation of IE Notice 87-66, 88-27, 88-45 and 88-98, it was noticed that 3 notices were closed without evaluation and 1 was lost.

- Evaluate the addition of 10 CFR 21 confirmation to receipt inspection requirements where applicable.

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

2. TMI-1 Liquid Radwaste SSFI - TDR No. 994 - May 31 to June 23, 1989

The overall objective was to determine if 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.

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 Stations," Rev. 0. The overall methodology for performing the inspection was as follows:

- research design bases information for the system;
- research work methods and practices which impact design bases of the system (i.e., modification, 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 all used during the inspection.

The overall conclusion reached was that the Liquid Radwaste System is expected to function as designed during normal operations and accident or transient conditions. Specific conclusions were:

1. The system's current functional requirements are articulated as design bases.
2. With the current system design, current functional performance requirements are met.
3. The system's current functional performance requirements have been preserved during modification. (There are two potential exceptions to this conclusion).
4. The operating procedures cause the system to be operated in accordance with the intent of the system's design.
5. The system is both tested and maintained to confirm and preserve the system's current functional performance requirements.
6. The training for operators, coupled with operating procedures, causes them to focus on and understand what the system is designed to do and how to recognize that it does what it is designed to do.
7. The licensing basis and subsequent revisions have been properly incorporated into system documents.

The strengths identified were:

- Procedures which govern operations and the administration of the Surveillance Test Program, with some notable exceptions, were clear, complete, and accomplished their objectives.

- Job Ticket Packages prepared by the Work Planning & Scheduling Department are both comprehensive and of good quality.

- Station personnel's' routine efforts to minimize the leakage of plant systems and control/restrict the use of chemicals used in the plant has a very positive effect on the overall liquid waste processing activity.

- Overall implementation and administrative control of the Environmental Qualification (EQ) Program, including the element of EQ spare parts identification, is meritorious.

- Classroom and "On-the-Job" Training.

Areas, where improvements would enhance confidence in system performance,

- Retrievability of up-to-date design bases information. The development of a current design basis document is currently in progress.

- Documentation of modifications.

- Development of a Post Maintenance Testing station administrative procedure. The development of a Post Maintenance Testing station administrative procedure is currently in progress.

- Performance of root cause analyses for failed components.

Thirty-one open items were reported for further evaluation. Note that some of these items may have been evaluated as requiring no further action. All of these open items have been evaluated and resolved.

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

3. TMI-1 High Pressure Injection System SSFI - October 22 to November 16, 1990

The overall objective was to assess whether the system has been modified, operated, maintained, and tested such that it would perform as designed during accident or abnormal transient conditions.

The governing document for this inspection was GPU Nuclear Specification No. 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 methodology used to perform the inspection consisted of:

- researching design bases information applicable to the system;
- researching work methods and practices which impact design bases of the system (e.g., modification, operations, maintenance, testing, procurement, and training); and,
- reviewing and evaluating 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 all used during the course of the inspection.

Difficulties were encountered in reconstructing both the design and licensing bases of the system. A significant number of man-hours were expended in efforts to retrieve historical data. A significant volume of historical design and licensing basis documentation was retrieved, completed and reviewed during the course of the inspection.

Several strengths were identified. These were as follows:

- Probabilistic Risk Assessment recommendations have been comprehensively addressed via implementation of modifications and revisions to operating procedures.
- The methodology of the Reliability Centered Maintenance Program is technically sound and its scope is comprehensive. Once fully implemented, it will increase confidence with respect to the operational readiness of the ECCS.
- The frequent verification of both mechanical and electrical system equipment line-ups by Operations ensures the operational readiness of the ECCS.
- Operator training is of good quality. Training required as a result of plant modifications is conducted in a timely and effective manner.
- Station personnel take positive action with respect to identifying and correcting adverse plant materiel conditions.

Several areas, where improvements would enhance confidence in system performance, were identified. These were as follows:

- The element of "instrument accuracy" is not consistently addressed in applicable engineering calculations; nor is it considered in surveillance test acceptance criteria.

- The methodology used to organize and file data in CARIRS appears to be undefined. As such, surveillance test results data, used to trend instrument performance, can not be readily accessed.

- Historical design and licensing basis information is difficult to retrieve. This appears to result in the inconsistencies that exist in some engineering evaluations that have been performed on a specific portion of the system during different times.

- Post maintenance testing requirements, as applicable to ECCS pump rotor replacements, warrants improvement. Post maintenance testing should verify that pump performance characteristics bound accident analysis assumptions over the entire operating range.

- There is presently no administrative mechanism in place to ensure that EP&I addresses the impact on diesel generator load capacity following safety-related pump rotor replacements. Rotor replacements alter pump runout flow power requirement.

- The specific identification of regulatory requirements and/or commitments is not reflected (by reference) into procedures and training materials. This can potentially create "continuing compliance" problems.

Several of the observations have been categorized as programmatic findings.

The overall conclusion reached is that the HPI System is expected to function as designed during accident or abnormal transient conditions. This overall conclusion is supplemented by the determination that GPU Nuclear continues to improve its established programs and the quality of the procedures used to implement these programs. Furthermore, it was concluded that GPU Nuclear has appropriately dispositioned previous SSFI findings via the Technical Functions Assigned Action Item (TFAAI) Program.

Thirty-eight open items were reported for further evaluation. These included one preliminary safety concern and one licensing action item. Note that some of these items may have been evaluated as requiring no further action. Of these 38 open items, two remain open. Both are associated with reviewing piping analyses for accuracy and/or need for revision.

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.

4. TMI - Unit No. 1 Low Pressure Injection System SSFI - July 20 to August 14, 1992

The primary overall objective was to determine if the system has been modified, operated, maintained, and tested in a manner that is consistent with its design basis. A secondary objective was to verify and validate design bases information contained in the System Design Basis Document.

The governing document for this inspection was GPU Nuclear Specification No. SP-9000-56-009, Rev. 3, "Technical Specification for Safety System Functional Inspection of Three Mile Island Unit 1 and Oyster Creek Nuclear Generating Station." The overall method used to perform the inspection consisted of:

- reviewing and evaluating the system design basis information presented in SDBD-T1-212 (Rev. 0), "System Design Basis Document For Decay Heat Removal System";
- evaluating the various programs and procedures that are used to verify, maintain or preserve the design bases of the system (e.g., engineering, modification, operations, maintenance, testing, and training); and,
- performing system walkdowns; conducting personnel interviews; and observing large break LOCA (LBLOCA) scenario simulator training exercises.

A concern, which constitutes a reservation regarding system operability as applicable to the injection mode of system operation, was identified. This reservation is manifested in an issue regarding emergency diesel generator loading under LBLOCA/LOOP conditions. GPU Nuclear had developed Preliminary Safety Concern (PSC) 92-003 to address this issue. Several individual concerns which, as a whole, constitute a reservation regarding system operability as applicable to the recirculation mode of system operation were also identified. This overall reservation is manifested in those issues specifically identified below.

The inspection team concluded that the LPI System is expected to function as designed during a LBLOCA provided that GPU Nuclear evaluates and appropriately dispositions the issues identified in this report.

Several strengths were identified. These were as follows:

- Recent development of the Reliability Centered Maintenance Program module applicable to the Decay Heat System has identified the need to implement twelve (12) additional preventive maintenance activities. Once implemented, they will enhance the operational readiness of the system's emergency safeguards function.
- During system walkdowns, component labeling was found to be both comprehensive and accurate.

- EP&I's planned method of evaluating the thermal overload protection for MOVs in the scope of the Generic Letter 89-10 Program was determined to be technically sound. This method will use the criteria of IEEE Standard 741.

- Operations Department personnel took prompt action to correct adverse plant materiel conditions identified during system walkdowns.

- The System Design Basis Document was determined to be of good quality. Its full benefit will be realized only if it is periodically updated to incorporate relevant design documentation as it becomes available. Recommendations to enhance this document are presented in various inspection observations.

A concern which constituted a reservation regarding system operability as applicable to the injection mode of system operation was identified. This reservation was manifested in the following issue:

- TDR-836 (Rev. 3), "Evaluation of Loading for the Emergency Diesel Generator and Engineered Safeguards (ES) Buses," does not accurately reflect D/G loading during LBLOCA/LOOP conditions.

Several individual concerns which, as a whole, constitute a reservation regarding system operability as applicable to long term core cooling (i.e., recirculation mode of system operation), were identified. This overall reservation is manifested in the following issues:

- Nonconservative reactor building pressure and sump fluid temperature values used in LPI pump net positive suction head calculation.

- Potential for vortex formation during and following suction swap-over from the borated water storage tank (BWST) to reactor building sump.

- Single failure vulnerability of BWST Lo-Lo level alarm instrument loop.

- Potential inability to remotely vent decay heat (DH) pumps when in recirculation mode of system operation.

- Single failure vulnerability of both long term circulation modes used to prevent boron concentration effects.

- Inability to access and manually operate valves DH-V19A&B and DH-V38A&B as required by abnormal transient procedure when in the recirculation mode of system operation due to post-accident radiation levels.

- Inadequate testing of DH pump discharge check valves DH-V16A&B.

Two issues regarding the relationship between the DH system and the LOCA reactor building temperature and pressure profile were identified. The issues consist of incorrect BWST "useable" volumes used in calculations. The impact of the error on the LOCA reactor building temperature and pressure profile was not determined, because the calculations employ the use of the CONTEMPT computer code.

Piping analysis ME-10 was found to contain several critical errors. As such, it did not adequately demonstrate that the piping will remain functional during a seismic event.

Various issues regarding the potential for DH system leakage and its resultant impact on both onsite and off-site dose were identified. Furthermore, the Technical Specification maximum allowable DH system leakage rate of 6 gph could not be verified to be correct.

It was determined that the LPI system has not been maintained in a configuration consistent with its original design basis. Injection line valves DH-V19A&B are maintained in a locked open position. Their design bases function is such that they be maintained in a locked throttled emergency safeguards position.

GPU Nuclear performed preliminary evaluations to determine the safety significance of each of the above observations.

Seventy open items were reported for further evaluation. Two of these items were addressed by one preliminary safety concern. Twenty-two of these items were evaluated as not requiring any further actions. Of these remaining 48 items, five of these items remain open. The fact that these items remain open has been recently addressed by NRC Inspection Report 50-289/96-201.

This SSFI was performed after System Design Basis Document SDBD-TI-212 was completed.

5. TMI-1 Service Water System Operational Performance Inspection - April 17-May 19, 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 inspection plan and scope were reviewed by NRC and concurred with in NRC letter to GPU Nuclear dated April 5, 1995. The self assessment inspection fulfilled all elements of the TI, and therefore adequately assessed the defined areas of inspection which included mechanical and electrical systems design, operations, maintenance, surveillance and testing, quality assurance, and corrective action. This service water system inspection involved the nuclear services river water (NR) and closed cooling water systems (NS), the decay heat river water (DR) and closed cooling water (DC) systems, and the reactor building emergency cooling water (RR) system. The SWSOPI assessed the operational readiness of these systems using vertical slice review techniques.

The objectives of the SWSOPI were to:

1. Assess the actions implemented in response to NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment,"

2. Verify the SWS are capable of fulfilling their thermal and hydraulic performance requirements. and

3. Assess the operational controls, maintenance, surveillance, and training related to the SWS.

The assessment was accomplished by performing a review of the above TMI-1 SWS components and system performance including design requirements, operating, maintenance, surveillance and other testing practices; maintenance and performance history; and quality assurance and implementation of corrective actions. Overall system operation was assessed to ensure it is being operated and maintained within its design bases to fulfill safety functions. Emphasis was placed on the evaluation of changes made to each system and the impact on the original design and licensing basis as well as any impact on related systems and programs.

The design review assessed the technical adequacy of each system concentrating on essential safety and functional characteristics. The review considered design conditions and transients, component classifications, equipment qualifications, 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. The design review included determinations of whether the system's design bases, design assumptions, calculations, analyses, boundary conditions, and models met licensing commitments and regulatory requirements. The inspection findings included specific concerns related to the ability of the SWS to perform their design bases functions, and the need to strengthen the link between the design bases and facility operations. Many of these findings were addressed, appropriate documentation has been updated, and it was satisfactorily concluded that the SWS systems meet their design bases and are capable of performing their safety function.

The operations area review evaluated the processes and procedures established to ensure the SWSs are operated in accordance with their design bases. Plant operating procedures, operator training, and plant configuration were assessed for adequacy and impact on SWS ability to perform its safety-related functions. The inspection identified specific instances where plant procedures did not provide guidance for all potential conditions and provided recommendations for strengthening these areas. Overall operation of the SWS was found to be adequate and operators' knowledge of SWS normal and emergency conditions was found to be thorough. Operator training, in general, adequately reflected the current plant configuration.

The maintenance evaluation verified that the maintenance performed on each system is

adequate to ensure that it will perform its safety-related function. Maintenance records were reviewed to determine if all system safety-related components are addressed by the maintenance program. The adequacy of programs and processes utilized for equipment failure trending, corrective action implementation, and root cause analysis was assessed. The adequacy of post-maintenance testing programs was evaluated. Inspection findings included recommendations related to additional heat exchanger inspection and cleaning, and inspection of infrequently used lines. Additional evaluation performed in response to a specific inspection concern verified that the effects of a certain corrosion inhibitor utilized had not adversely affected the design bases capability of the SWS. In general, maintenance practices and procedures and post-maintenance testing were determined to be adequate. Material condition for the SWS and structures was acceptable.

The surveillance and testing portion of the review verified that testing performed on each system is sufficient to demonstrate the capability to perform its intended functions during the most severe operating conditions. The evaluation focused on functional testing of SWS systems and components. The inspection identified that Technical Specification surveillance testing was adequately defined and properly implemented. Overall, the IST Program was found to be well defined and in accordance with regulatory requirements. Trending of IST data was found to be effectively monitoring component performance.

Quality assurance program activities including corrective action, operability determinations, trending and quality verification were evaluated to determine the adequacy of these programs in identifying and correcting deficiencies. Some examples were identified where corrective actions were not identified or not fully effective. These findings have served to strengthen the corrective action processes. Overall, the inspection identified that actions to address immediate operational problems were timely and effective.

A total of eighty six (86) Licensing Action Requests were issued as part of the GPU Nuclear Licensing Information Tracking System to track identified open items. As of February 10, 1997, fifty nine (59) items have been closed.

The overall results of this self-assessment inspection support the conclusion that SWS configuration control and performance are consistent with the design bases and that design bases requirements in general have been adequately translated into operating, maintenance and testing procedures. Inspection findings noted above have strengthened existing programs established to control and maintain design bases requirements.

Attachment 4

NRC Vertical Slice Program Review/Inspections

This attachment provides details on the following NRC Inspections:

1. TMI-1 NRC Performance Appraisal Team (PAT)/SSFI Inspection
2. TMI-1 NRC Performance Appraisal Team (PAT) Inspection
3. TMI-1 NRC Electrical Distribution System Functional Inspection

NRC Vertical Slice Program Review/Inspections

The following are summaries of NRC inspections which looked at system design bases adequacy and implementation and/or programmatic requirements regarding design control.

1. TMI-1 NRC Performance Appraisal Team (PAT)/SSFI Inspection - March 3-27, 1986 (Inspection Report 86-03).

This NRC inspection was conducted by the members of the Performance Appraisal Section from the Office of Inspection and Enforcement, Region I inspectors, and NRC contractors. The inspection was performed in response to Commission Memorandum & Order CLI-85-09 which lifted the 1979 TMI-1 shutdown order. This comprehensive inspection assessed the operational readiness and functionality of the TMI-1 emergency feedwater (EFW) system, as well as the operational readiness of systems supporting the EFW system, such as the 250 Vdc station batteries, the two-hour backup supply air system, and the integrated control system (ICS). Operational readiness and management controls were reviewed in the following six (6) functional areas, primarily as they related to the EFW system:

- Design Changes and Modifications
- Maintenance
- Surveillance Testing
- Operations
- Quality Assurance
- Training

The inspection objectives included determining whether:

1. the system was capable of performing the design basis safety functions,
2. testing was adequate to demonstrate that the system would perform the required safety functions,
3. system maintenance was adequate to ensure system operability,
4. operator and maintenance training was adequate to ensure proper operation and system maintenance, and
5. that human factors considerations relating to EFW system and supporting system procedures were adequate to ensure proper operation under normal operating conditions.

The inspection identified weaknesses in the performance of activities related to design changes

and plant modifications and recognized that this area had not achieved the high standards attained in the other areas inspected. The results of this inspection provided valuable insights as to where additional management attention and involvement needed to be focused. The follow-on NRC PAT inspection 86-14 recognized that considerable effort had been made to improve plant modification programs and governing procedures, and that training of responsible personnel had been conducted on the new design change process procedures and the overall regulatory requirements for design control.

The inspection report noted strong performance in the areas of training, quality assurance, maintenance, and operations.

Management commitment to training at TMI-1 was identified as a strength. Operator requalification training programs were found to incorporate significant plant modifications and were taught prior to startup. Information on minor modifications was effectively provided to operating shifts by training letters. Operators, in general, were determined to be knowledgeable concerning system operating procedures, failure modes, and emergency procedures.

EFW system surveillance test procedures were found to be generally adequate for demonstrating system functionality, thus reflecting effective translation of system design bases requirements into testing procedures.

Preventive maintenance was identified as a strength. Extensive trending based on corrective maintenance activities was noted as a strength and effectively identified problem areas on both a system and component basis. Preventive maintenance activities conducted on the non-safety related ICS, supporting EFW, was considered extensive. These results provided assurance that design bases requirements are adequately translated into maintenance procedures.

Quality assurance programs were found to be generally strong and effective. The GPU Nuclear QA organization was noted as being fully capable of identifying in-depth technical and design bases issues, and monitors had the necessary training and experience to identify design and design control problems. The inspection found efficient and effective systems in place to track and provide management review of actions required to correct deficiencies identified by the QA organization and other sources.

Additionally, it was determined that the independent technical and safety process activities and procedures satisfied Technical Specification requirements and that training of safety reviewers was satisfactory. The effectiveness of these independent review functions provides assurance that design bases compliance is maintained when facility changes are implemented.

2. **TMI-1 NRC Performance Appraisal Team (PAT) Inspection - August 25-September 5, 1986 (Inspection Report 86-14).**

This NRC inspection was conducted by the NRC Performance Appraisal Section from the

Office of Inspection and Enforcement and Region I inspectors, and was performed in response to NRC Commission Order CLI-85-09 which lifted the 1979 TMI-1 shutdown order. This inspection concentrated on the evaluation of management controls in five (5) functional areas:

- Operations
- Design Changes and Modifications
- Maintenance
- Safety Review Activities
- Surveillance Testing

This NRC inspection did not concentrate on any one system, but assessed the management controls in each area as applied to licensed activities for all plant systems. Each area was reviewed to determine the adequacy of established programs and the effectiveness of program implementation. As such, this inspection provides a measure of the adequacy and effectiveness of programs and processes in place to ensure configuration control, compliance with design bases requirements, and translating design bases requirements into operating, maintenance, and testing procedures. The inspection objectives specifically addressed whether:

1. written policies, procedures, or instructions to provide management controls in the subject area were provided;

2. policies, procedures, and instructions were adequate to ensure compliance with regulatory and internal requirements;

3. personnel understood their responsibilities and were adequately qualified, trained, and retrained to perform their responsibilities; and

4. requirements of the subject area had been implemented and appropriately documented in accordance with management policy.

The NRC inspection team determined that the management controls for licensed activities in the five (5) functional areas inspected at TMI-1 were generally adequate and strengths were identified in the functional areas of Operations, Maintenance, and Surveillance Testing.

Specifically, the inspection found operating procedures were generally technically adequate and design bases requirements are adequately translated into such procedures. The inspection also determined that Technical Specification surveillance test programs, overall, were a strength. Test procedures reviewed were technically adequate and test results were reviewed on a timely basis. These results provide assurance that design bases requirements are adequately translated into operating and testing procedures.

Configuration control programs were reviewed for accomplishing major plant modifications as well as the expedited processes for accomplishing minor modifications. These reviews were

performed with emphasis on the design change program and processes. The inspection identified that plant modification programs and governing procedures had been enhanced based on earlier identified weaknesses and that these programs and procedures adequately comply with regulatory commitments. The report further identified that modifications selected for minor modification processes were appropriately selected, and the design analyses and safety reviews were adequately implemented for these modifications. These results provide reasonable assurance that configuration control programs and processes are appropriately considering and maintaining plant design bases compliance.

The inspection identified that corrective maintenance procedures were determined to be adequate and corrective maintenance programs were generally well managed. Vendor technical manual recommendations were found to be incorporated into procedures and drawings, and manuals were updated when necessary. Post-maintenance testing was found to be adequately conducted. Additionally, preventive maintenance programs overall were identified as a strength; thus providing further assurance that design bases requirements are adequately translated into operating, maintenance, and testing procedures.

Although initial weaknesses were identified in the safety review process implementation, these have been addressed and safety review process procedures and training programs have been upgraded. Subsequent inspections, as well as internal safety review process assessments, have shown that these corrective actions have been effectively implemented and the safety review process procedures provide adequate assurance that design bases requirements are maintained.

Material non-conformance dispositions reviewed were determined to be adequate and well documented, thus providing a measure of the effectiveness of established corrective action programs.

3. TMI-1 NRC Electrical Distribution System Functional Inspection - November 19, 1990 - December 21, 1990 (Inspection Report 90-81)

This NRC inspection was conducted by NRC Region I, NRR, and NRC contractor personnel to determine if the electrical distribution system (EDS) was capable of performing its intended safety functions, as designed, installed and configured. A second objective of the inspection was the assessment of GPU Nuclear engineering and technical support of the EDS activities.

The inspection reviewed calculations and design documents and focused on those attributes which ensure that adequate power is delivered to these systems and components that are relied upon to remain functional during and following a design basis event. The review covered portions of onsite and offsite power sources and included the 230KV offsite power grid, station auxiliary transformers, 6.9 KV system, 4.16 KV Class 1E system, emergency diesel generators, 480Vac Class 1E unit substations and motor control centers, station batteries, battery chargers, inverters, 250/125 Vdc Class 1E busses, and the 120 Vac Class 1E

vital distribution system. Regulation of power to essential loads, protection for calculated fault currents, circuit independence, and coordination of protective devices were also reviewed to verify the adequacy of the emergency onsite and offsite power sources.

This inspection also assessed the adequacy of mechanical systems which interface with and support the EDS including air start, lube oil, and cooling systems for the EDGs, and cooling and heating systems for the electrical distribution equipment.

The inspection team also reviewed the adequacy of procedures and guidelines governing design calculations, design control, and plant modifications, and assessed the effectiveness of the controls established to ensure that the design bases of the EDS are maintained.

As-built configuration was verified to be consistent with the design documents and modification packages. Additional attributes of plant modification processes evaluated included the design review process and 10 CFR 50.59 safety evaluations.

The inspection reviewed conformance to GDC 17 and 18 and appropriate criteria of Appendix B to 10 CFR 50. The plant Technical Specifications, FSAR and appropriate safety evaluation reports were reviewed to ensure that technical requirements and commitments were being met.

EDS equipment was examined to verify configuration and ratings which included original installations, as well as equipment installed through modifications.

The inspection team also reviewed maintenance, calibration, and surveillance activities for selected EDS components.

The inspection concluded that GPU Nuclear had established adequate configuration control for most electrical equipment and, in general, adequate criteria had been established for surveillance testing of the EDS equipment. Inadequate design control was observed for one specific modification, which indicated the need for increased attention to the modification process. This specific case has been corrected. Additional weaknesses identified were primarily in the area of available documentation particularly in original design bases documents. Design bases documents for the EDS were not available at the time of the inspection.

Specific strengths were identified which represent the quality of engineering and technical support. These strengths included proper and timely dispositioning of material non-conformances and licensee event reports, thorough investigation of root causes, and detailed engineering modifications and design changes. Maintenance was found to be adequate with effective setpoint control and calibration of protective devices. Maintenance and test procedures were found to be acceptable and technically adequate, and test and calibration records reviewed ensured operation within applicable acceptance ranges. Programs for controlling and implementing temporary modifications and bypasses in the plant were

determined to be acceptably maintaining design bases compliance.

Self-assessment programs and quality assurance activities were determined to be adequately monitoring engineering effectiveness.

Personnel were found to have received adequate training required for performing engineering and design functions and for providing technical support for plant operational activities. Engineering and technical support was identified as a strength and provided a basis for the safe operation of the plant.

Overall, the inspection determined that, generally, the TMI EDS is capable of performing its intended safety function. With the exception of specific identified findings, EDS equipment and components were determined to be adequately sized and configured, although some margins were found to be minimal. It was concluded that design implementation of the EDS at TMI-1 was acceptable based upon the sample of design drawings, studies and calculations reviewed and equipment inspected. The results of the inspection provide reasonable assurance that design bases requirements are adequately translated in operating, maintenance, and testing procedures, and that system configuration and performance are consistent with the design bases.

Attachment 5

Problem Identification and Corrective Action Processes

This attachment provides a detailed description of problem identification and corrective action processes for TMI-1.

Problem Identification and Corrective Action Processes

Description

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 all workers have the right and responsibility to identify and report safety concerns

These policies are implemented by plans and procedures. The quality assurance program establishes the following general requirements concerning nonconforming conditions and

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[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

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[7]
- Stop Work Notification[8]
- Shutdown Recommendation/Directive[9]
- GPU Nuclear Corrective Action Programs and Processes[10]
- NRC Regulation 10-CFR-21 (Reporting of Defects and Noncompliance)[11]
- GPU Nuclear Quality Deficiency Reports (QDR)[12]
- Event Review and Reporting Requirements[13]
- Material Nonconformance Reports and Receipt Deficiency Notices[14]
- Nuclear Safety Assessment (NSA) Audit Program[15]
- Management Escalation Program for NSA Quality Deficiencies[16]

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[7] GPU Nuclear Corporate Policy and Procedure Manual, 1000-PLN-1291.01, Revision 6
[8] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7202.01, Revision 2
[9] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7202.02, Revision 2
[10] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7216.01, Revision 1
[11] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-1290.01, Revision 5
[12] GPU Nuclear Corporate Policy and Procedure Manual, 1000-ADM-7215.02, Revision 5
[13] Three Mile Island Administrative Procedure 1044, Revision 34
[14] Three Mile Island Administrative Procedure 1077, Revision 2
[15] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7218.01, Revision 1
[16] Nuclear Safety Assessment Department Procedure Manual, 1110-ADM-7216.01, Revision 0

- Incident Critique Procedure[17]
- NSA Monitoring Program[18]
- Deficiency Trend Analysis[19]
- NSA Vendor Surveillance[20]
- Independent Onsite Safety Review Group Procedure[21]
- Safety Issues Assessment Program[22]
- OQA Plan Document Reviews[23]
- Supplier Corrective Action Request[24]
- Access Authorization and Fitness-For-Duty Program Audits of Contractors[25]
- NSA Vendor Audit Program[26]

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 TMI-1. 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 TMI-1'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 a material nonconformance is identified, a Material Nonconformance Report (MNCR) is issued to ensure the material is identified and controlled to prevent inadvertent use. This process includes determinations of reportability, equipment operability and the cause of the nonconformance. MNCRs are trended to detect repetitive problems and significant deficiencies.

When significant problems are identified they are entered into an approved corrective action system such as the Quality Deficiency Report (QDR) or Licensee Event Report (LER). A QDR can be initiated by any individual identifying a quality deficiency. The QDR is reviewed by the

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

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.

For lower threshold issues the Event Capture and Reporting process is used. This process serves to reasonably assure that events and situations which may require further review, reporting, or corrective action are identified, controlled, documented, reported, and evaluated. A senior management review group reviews the event and assigns the issue to a responsible department. The responsible department determines the event category based on consequences or potential consequences of the event, evaluates whether an approved corrective action system is applicable and assigns an action party. The three event categories are: analyzed event, awareness event or trended event. The analyzed events are of sufficient consequence, near consequence, or management interest to warrant a systematic root cause evaluation. The awareness events are such that a less rigorous root cause analysis may be sufficient. The trended events are events which warrant trending, but no formal evaluation or management response. The action party reviews and resolves the event using root cause analysis techniques when warranted. A Human Performance Review Board, the senior management group responsible for assessing the results of the process, periodically reviews the event data to identify performance trends. They also evaluate the appropriateness and effectiveness of the corrective actions.

Irrespective of the reporting system (LER or QDR) for significant problems, root cause is determined. More than 50 personnel have received training in techniques such as, NRC Human Performance Investigation Process, Human Performance Evaluation System, 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. The GORB reviews and reports annually on the effectiveness of the independent oversight activities of the NSA department.

Future Plans

Efforts are underway to enhance the current corrective action processes. A team has identified and incorporated best industry practices concerning corrective action into GPU Nuclear processes at TMI-1. Implementation of the revised procedures is planned to begin at TMI-1 in March 1997, following training. The enhancements improve and simplify the problem identification mechanism. 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 Three Mile Island Nuclear Station Unit 1.

Commitments

The following table provides a list of all commitments being made in this response by GPU Nuclear for the Three Mile Island Unit 1 Nuclear Station.

Commitment Number	Description	Completion Date
1	Develop Engineering Division Procedure EP-045 to improve the review and update processes 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 of SDBD and SSFI open items by entering them into the Corrective Action Program 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	Conduct personnel training on design bases issues and the new FSAR update process (EP-045).	12/31/97
5	Develop a specific plan and schedule for improving setpoint basis documentation/references.	5/30/97

SDBD = System Design Basis Document
TDR = Technical Data Report

Attachment 7

TMI-1 Design Bases Reconstitution Program

This attachment provides a detailed description of GPU Nuclear's TMI-1 Design Bases Reconstitution Program.

TMI-1 Design Bases Reconstitution Program

I. Summary

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

1. providing insights into the fundamental design philosophies and intents;
2. identifying gaps in the design documents, design process documents and design information;
3. identifying changes to operating, maintenance and testing procedures.

The results derived from conducting the TMI-1 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 B&W Owners Group reconstitution project and subsequent reproduction of original B&W engineering design bases information was completed.

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 the design information and the collation of that information in the Design Basis Document. That program added the Design Basis Document to the 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, maintained and altered conformed to the Design Basis Document.

GPU Nuclear did have design bases information, including 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 bases 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 bases 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 bases information.

2. GPU Nuclear System Engineering Function

The system engineer/owner function is intended to provide or coordinate all 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 information for the selected systems as possible; capture in one place 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 Bases Reconstitution Program and to formulate the schedule for completing the program has changed over time 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 program;
- plant life extension/license renewal; and,
- conduct of Safety System Functional Inspections or equivalent vertical slice system assessments;

Criteria used to determine that design bases reconstitution was not necessary included:

- engineering design bases was perceived to be readily reconstructable; or,
- the change in the engineering design bases was relatively recent and therefore perceived to be accessible.

3. Role - GPU Nuclear Engineers

From 1989 through 1993, the GPU Nuclear Design Bases 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 bases documentation (e.g. B&W 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 bases information;
- availability of capable engineers who could prepare design bases 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 reference to the source documents and other design documents for selected systems.

5. Recovery of Babcock & Wilcox (B&W) Design Information

The B&W Owner's Group formed a Configuration Management Project (B&WOG EB-90-12A). This Project was chartered in 1989 to address generic design bases records for the original BWNT scope of supply.

The project consisted of three phases:

1. identification of generic key parameters that should be included in the NSSS Design Base; definition and development of the index format to be used in phase 2;
2. development of a categorical index of current B&W Nuclear Service Company (BWNS) files and documents which is accessible, maintained and contains appropriate cross-references; and,
3. identification of gaps in the NSSS Design Base between the parameters developed in phase 2 and key parameters listed in phase 1.

The project has been completed. GPU Nuclear has received the following:

- "Configuration Management Database Structure" (47-1178193-00, February 1990);
- "Key Plant Parameters for the 177 Fuel Assembly Design" (47-1178198-00, January 1990);
- Boxes of hard copies of design basis and associated supporting design documentation; and,
- the CONFIG database organized around key parameters containing approximately 20,000 records.

6. Recovery of Gilbert-Commonwealth/Parsons Design Information

Gilbert-Commonwealth/Parsons has been continuously involved in the GPU Nuclear Design Bases Reconstitution Program since 1989. They have performed research of the design documents and supporting design information which they retain, defined the function why's, performed interviews of original designers and

prepared SDBDs.

Subsequent to the formation of a TMI-1 Design Bases Reconstitution group in 1994, Gilbert/Parsons was retained to continue to perform document research and interviews of original designers. In addition, Gilbert/Parsons provided technical reviews for concurrence of SDBDs prepared by GPU Nuclear personnel.

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 reevaluated the current status of its Design Bases Reconstitution Program in August 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 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.

B. Future Activities

GPU Nuclear will continue to assess and define the future scope and output of its Design Bases Reconstitution Program. The Design Bases Reconstitution effort currently in progress is provided by Enclosure 2. The systems listed in Enclosure 3 as under consideration are those that currently require additional evaluation to determine if there is a need to complete a design bases reconstitution.

ENCLOSURE 1

Three Mile Island System Design Bases Reconstitution Activities -Completed

A. Design Bases Documents

Twenty-one Design Basis Documents covering thirty-five systems have been completed and released for use. These Design Basis Documents are:

1. SDBD-TI-211; "Makeup and Purification System" Revision 1 (11/12/96).

2. SDBD-TI-212; "Decay Heat Removal" Revision 0 (7/14/92), Revision 1 (12/5/94).

3. SDBD-T1-213; "Core Flooding System" Revision 0. (7/5/96)

4. SDBD-TI-214; "Reactor Building Spray System"; Revision 0 (3/24/92); Revision 1 (3/17/94).

5. SDBD-TI-220; "Reactor Coolant System"; Revision 0 (12/20/90).

6. SDBD-TI-232; "Liquid Radioactive Waste System"; Revision 0 (11/30/89).

7. SDBD-TI-411; "Main Steam System Turbine Bypass System, Heated Post Support System"; Revision 0 (8/1/90).

8. SDBD-TI-413; "Extraction Steam System"; Revision 0 (12/29/92), Revision 1 (9/2/93).

9. SDBD-TI-420; "Condensate/Feedwater System"; Revision 0 (7/9/92).

10. SDBD-TI-431/432; "Heater Drain System, Heater Vent System, Moisture Separator Drain System"; Revision 0 (4/29/93), Revision 1 (10/28/94).

11. SDBD-TI-531; "Nuclear Services River Water System, Nuclear Services Closed Cooling Water System, Intermediate Closed Cooling Water System"; Revision 0 (1/31/96).

12. SDBD-TI-533; "Decay Heat River Water, Decay Heat Closed Cooling Water"; Revision 0 (3/6/96).

13. SDBD-TI-534; "Reactor Building Emergency Cooling Water System", Revision 0 (9/20/89).

14. SDBD-TI-544; "Secondary Services Closed Cooling Water System, Secondary Services River Water System, Mechanical Draft Cooling Towers, Screen Wash and Sluice System, River Water Pump Lubrication System"; Revision 0 (12/20/95).

15. SDBD-TI-622; "Control Rod Drive Control System, Control Rod Drive Mechanism"; Revision 0 (12/17/94).

16. SDBD-TI-641; "Reactor Protection System"; Revision 0 (12/21/90).

17. SDBD-TI-642; "Engineered Safeguards Actuation System"; Revision 0 (1/8/91); Revision 1 (5/20/94).

18. SDBD-TI-661; "Radiation Monitoring"; Revision 0 (12/20/91).

19. SDBD-TI-823; "Reactor Building Cooling System"; Revision 0 (7/13/94).

20. SDBD-TI-826/827; "Control Building Ventilation System/Automatic Temperature Control, Compressed Air System, Chilled Water System"; Revision 0 (9/7/90), Revision 1 (2/8/91), Revision 2 (6/14/94).

21. SDBD-TI-852; "Instrument Air System"; Revision 0 (12/21/92).

B. GPU Nuclear Self Assessments

Five self assessments consisting of four Safety System Functional Inspections and one Operational Performance Inspection have been completed. These are:

1. TDR 992; Emergency Electric Power Distribution; 1/29/90.

2. TDR 994; Liquid Radwaste System; 11/30/89.

3. TDR 1034; High Pressure Injection System; 12/14/92.

4. TDR 1092; Low Pressure Injection System; 1/12/93.

5. TDR 1166; Service Water Systems; 7/26/95.

ENCLOSURE 2
Three Mile Island Design Basis Reconstitution Activities - In Progress

1. A design bases reconstitution of electrical systems which will include the following electrical subsystems:

   - 4160 Volt Auxiliary System
   - 120 Volt AC Vital Power
   - 120 Volt AC Regulated Power
   - 480 Volt Auxiliary System

   This reconstitution effort consists of establishing an organized listing and collection of design bases information, source documents and references.

2. A design bases reconstitution of additional systems is also in progress:

   - In-core Monitoring System
   - Circulating Water System

ENCLOSURE 3
Three Mile Island System Design Basis Documents - Under Consideration

Design Basis Reconstitution
System Priority - Three Mile Island

System Name	SE Rank	PE Rank	ME Rank	EE Rank	PU Rank	Priority Rank	Action
Emergency Diesel Generator (RSS)	1		1	2	1	1	UC
Spent Fuel Pool Cooling (RSS)	1		2		1	1	UC
Emergency Feedwater (RSS)	1		1		1	1	UC
250/125 Volt DC (RSS)	1	1		2		1	UC
Cable Routing	1	2		3		2	NTBD
Radwaste Gas	1		2			2	NTBD
Nuclear Instrumentation	1		2		2	2	NTBD
Cable Separation	1			2		2	NTBD
Containment Bldg. Pene	1	2	2	2		2	NTBD
Fire Protection	1	2	2	4	2	2	UC
Reactor Bldg. & Leak Rate	1		3		2	2	NTBD
Pressurizer	1		3		1	2	NTBD
Reactor Coolant Pressure Control	1					2	NTBD
Heat Sink Protection System (RSS)	1					2	UC
Rules for Analysis			1			2	NTBD
Natural Draft Cooling Twr.		2				3	NTBD
6.9 KVA Auxiliary System (RSS)		2		4		3	UC
Security Lighting		2		3		3	NTBD
Raceway Supports		2		3		3	NTBD
230 KVA Main Step-down XFMRs (RSS)				4	2	3	UC
Main Generator				4	2	3	NTBD
Main Generator Excitation				4	2	3	NTBD
Pressurizer Level Control						4	NTBD
Pressurizer Heaters						4	NTBD
Main and Auxiliary XFMRs (RSS)						4	UC
Electrical Heat Tracing				4		4	NTBD
Plant Lighting				4		4	NTBD
Chemical Addition			3			4	NTBD
Lightning Protection				4		4	NTBD
Cathodic Protection				4		4	NTBD
Water Treatment			3			4	NTBD
Chemistry			3			4	NTBD
Integrated Control System			3			4	NTBD
Misc. Non Vital Distribution				3		4	NTBD

LEGEND: (a) SE = System Engineering; (b) PE = Plant Engineering; (c) ME = E&D/Mechanical Engineering; (d) EE = E&D/Electrical Engineering; (e) PU = Power Uprate; (f) RSS = Maintenance Rule Risk Significant System; (g) UC = Under Consideration; (h) NTBD = Not to be done.

PRIORITY: (1) Important to have; (2) Should have; (3) Would like to have; (4) Optional.