Duquesne Light Company Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

JAMES E. CROSS
Senior Vice President and
Chief Nuclear Officer
Nuclear Power Division

(412) 393-5255
Fax (412) 643-8069

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to Request for Information Pursuant to 10 CFR 50.54(f)
Regarding Adequacy and Availability of Design Bases Information

Enclosed with this letter is the Duquesne Light Company (DLC) response to the Nuclear Regulatory Commission (NRC) October 9, 1996, request for information regarding the adequacy and availability of design bases information. The attached information provides DLC's current bases for concluding that there is reasonable assurance that the Beaver Valley Power Station, Units No. 1 and No. 2 (BVPS), are operated and maintained within the design bases and that deviations are identified and corrected in a timely manner.

DLC shares the NRC's concerns regarding consistency with the design bases. Accordingly, we place great importance on configuring, operating, and maintaining BVPS in accordance with its design bases. DLC recognizes and appreciates that compliance with the design bases is a key element for assuring public health and safety.

BVPS has acted on this concern by either taking or embarking on a number of initiatives that include the assessment of the processes for maintaining consistency with the design bases and taking corrective actions when deviations were recognized. Three processes of particular note are the vertical slice reviews called safety system functional evaluations, a recent re-engineering of the BVPS condition report program, and a detailed UFSAR review. The self-initiated safety system functional evaluations, started in 1988, identify and correct design deficiencies. Accordingly, they have been recognized by the NRC as forms of design reconstitution. Recently, management has re-engineered the processes for identifying problems, taking corrective action, and assessing their effectiveness. On December 26, 1996, DLC submitted a commitment to conduct a detailed review of BVPS Units No. 1 and No. 2 Updated Final Safety Analysis Reports. Details of these initiatives are discussed in the attached response.

1/1

A074

This response was developed by a team comprised of site personnel who have direct knowledge about the matters addressed. The response was independently reviewed by experienced personnel who are recognized as having substantial industry experience. The response preparation process was designed to provide me with a documented basis on which I could base a determination of reasonable assurance that BVPS Units No. 1 and No. 2 are operated and maintained within their design bases and that deviations are reconciled in a timely manner. An important part of that process was the inclusion of verification measures for all factual statements.

Recent plant events have resulted in additional management attention in areas dealing with vendor oversight and plant system alignment. DLC understands the significance of these events as demonstrated by its prompt and comprehensive corrective actions. DLC finds that the broader implications of the events are comparable to those associated with other situations discussed in the report.

A positive indicator of the increased awareness of plant personnel on the importance of complying with the UFSAR is the increased number of condition reports that have been written for discrepancies between plant documentation and the UFSAR. This information was used to develop the commitment to complete a detailed review of the UFSAR by December 31, 1998.

Based on the review of this document and the knowledge of the plant, it is concluded that consistency between the design bases, and operating, maintenance, testing procedures, program changes and design changes, is implemented by a group of mutually complementing procedures that have been created in accordance with regulatory requirements, including those of 10 CFR 50, Appendix B, and structured so as to provide reasonable assurance of continuing compliance. That consistency with the BVPS current design bases is demonstrated by this response, which provides reasonable assurance that Units No. 1 and No. 2 are configured and operated in accordance with their design.

If you have any questions regarding the information contained in this submittal, please contact Raymond A. Hruby, Jr., (412) 393-5705.

Sincerely,

James E Cross

James E. Cross

c: Mr. S. J. Collins, Director, NRC NRR
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. S. Brinkman, Sr. Project Manager

bc: ORC Members
OSC Members

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF BEAVER)

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Response to Request For Information Pursuant To 10 CFR 50.54(f)
Regarding Adequacy And Availability Of Design Bases Information

Before me, the undersigned notary public, in and for the County and Commonwealth aforesaid, this day personally appeared James E. Cross, to me known, who being duly sworn according to law, deposes and says that he is President, Generation Group, Duquesne Light Company, and Chief Nuclear Officer of the Beaver Valley Power Station, he is duly authorized to execute and file the foregoing submittal on behalf of said Company, and the statements set forth in the submittal are true and correct to the best of his knowledge, information and belief.

James E. Cross

James E. Cross

Subscribed and sworn to before me

on this 7th day of February, 1997

Tracey A. Baczek

Notary Public

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 1997
Member, Pennsylvania Association of Notaries

ATTACHMENT

Response to Request for Information Pursuant to 10 CFR 50.54(f)
Regarding Adequacy and Availability of Design Bases Information

Duquesne Light Company

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION

UNITS No. 1 & No. 2

RESPONSE TO

NRC REQUEST FOR INFORMATION

PURSUANT TO 10 CFR 50.54(f)

ADEQUACY AND AVAILABILITY OF

DESIGN BASES INFORMATION

February 7, 1997

TABLE OF CONTENTS

EXECUTIVE SUMMARY

This is the Duquesne Light Company (DLC) Response to the October 9, 1996 Nuclear Regulatory Commission (NRC) request for information pursuant to 10 CFR 50.54(f) regarding how the Beaver Valley Power Station (BVPS), Unit No. 1 (BVPS 1) and Unit No. 2 (BVPS 2) (each, a "Unit", and collectively, the "Units"), are operated and maintained within the design bases and how deviations from the design bases are reconciled in a timely manner. This Response provides information which gives reasonable assurance that the Units are so operated and maintained and that any such deviations are reconciled in a timely manner.

To prepare this Response, DLC management formed a multi-disciplined team of knowledgeable and experienced BVPS employees to formulate a response plan, collect supporting information, organize information in support of the requested information, assess the adequacy of the gathered information, and prepare the submittal to be responsive to the NRC's request. A team charter was developed containing a mission statement, goals and objectives, team membership, and a methodology for addressing the 10 CFR 50.54(f) questions. The Response Team worked closely with BVPS management in preparing this Response. Management participation and review provided additional assurance that the information contained herein is accurate and responsive to the NRC's request for information, and that this submittal represents the collective knowledge of those responsible for operating and maintaining BVPS.

Active overview of the actions taken to ensure that the information supporting this Response is complete and accurate was provided by the Quality Services Unit, Independent Safety Evaluation Group, and members of the Nuclear Safety Review Board. The preparation process and verification measures provided a framework for finding reasonable assurance that BVPS 1 and BVPS 2 have been and are operated and maintained within their design bases and that deviations are reconciled in a timely manner.

In its request for information, the NRC specifically enumerated the categories of information licensees are to provide. DLC has organized this Response in accordance with the order of the NRC's specific requests, as follows:

Section 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."

Section A of this Response describes the engineering design and configuration control processes at BVPS. It focuses on the DLC Operations Quality Assurance Program and the various mechanisms, procedures and processes that are in place to implement the Program's engineering design and configuration control aspects. The Section describes mechanisms used to document design bases information and to ensure consistency with the design bases when facility modifications are made. Various configuration control

processes also are discussed in Section A, including those implementing 10 CFR 50.59 safety evaluations, 10 CFR 50.71(e) Updated Final Safety Analysis (UFSAR) changes and license amendment requests, as well as a program of vertical slice reviews initiated in 1988 and modeled after the NRC's Safety System Functional Inspection methodology. The BVPS 1 vertical slice reviews included eight (8) Safety System Functional Evaluations (SSFEs); one (1) Service Water Operational Performance Inspection (SWOPI) of the River Water System (this inspection included BVPS 2 Service Water System as well); and one (1) Safety System Functional Inspection (SSFI) of the Safety Injection System. BVPS 1 has been the focus of these reviews because it was licensed prior to the Three Mile Island (TMI) incident, when licensing and pre-operational testing requirements were less stringent than those which were applied to BVPS 2, which was licensed after the TMI incident. BVPS 2 was required to engage in more rigorous pre-operational and start-up testing activities. Neverth ess, as part of its assessment for preparing this Response, DLC recently has conducted two (2) focused Vertical Slice Reviews at BVPS 2 to further assure itself of that Unit's consistency with its design bases.

The implementation and functioning of the mechanisms, programs, procedures and processes described in Section A of this Response provide additional confidence that applicable design requirements, regulations, codes and standards are correctly translated into appropriate design documents such as calculations, design analyses, specifications and drawings.

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

Section B of this Response provides DLC's rationale for concluding that design bases requirements have been properly translated into operating, maintenance and testing procedures. This rationale is based upon established and continually improved procedure change control processes and the results of routine and special initiative assessment programs.

When BVPS procedures were developed during the initial licensing of BVPS 1 and BVPS 2, they were based on pre-operational and start-up testing activities which demonstrated the Units' readiness and capability to operate safely, consistent with its design bases. Comprehensive change control processes have been in effect in accordance with 10 CFR 50, Appendix B. These processes are implemented in accordance with the DLC Operations Quality Assurance Program, and contain attributes to assure continuing consistency between procedures and design bases requirements. Controls discussed in Section B, such as general and specific procedural provisions, and multi-disciplinary review of procedures and design bases information by various organizations, are used to help ensure that design bases requirements are translated into procedures. Moreover, in recognition of the critical role played by personnel who are called upon to implement these processes, the procedure development and revision process is conducted by knowledgeable and trained staff.

Section B also discusses DLC's implementation of routine, ongoing programs, as well as special initiatives such as the vertical slice reviews, which verify that design bases requirements are appropriately reflected in procedures. As these programs, reviews and initiatives have identified deficiencies, corrective actions have been or are in the process of being implemented.

Collectively, these programs and special initiatives, together with internal and external audits, provide reasonable assurance that design bases requirements are appropriately translated into operating, maintenance and testing procedures.

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

DLC's rationale for concluding that the configuration and performance of structures, systems and components (SSCs) at BVPS are consistent with their design bases is discussed in Section C, which sets forth a number of factors supporting that rationale. First, a start-up testing program was developed and implemented at both BVPS 1 and BVPS 2 before each Unit began commercial operation, providing a benchmark or original basis for confidence that SSC configuration and performance for both Units are consistent with the Units' design bases. Since then, routine operation, maintenance and modification activities, and the implementation of controlled, comprehensive procedures and processes which control design changes and operating, maintenance, and testing activities, have helped to ensure such consistency.

Section C also describes configuration management, ongoing performance testing, inspections and other routine programs to verify that SSCs are maintained and perform within the BVPS design bases. Along with these, DLC has undertaken special initiatives at BVPS, also discussed in Section C, to provide ongoing confirmation that SSC configuration and performance are consistent with design bases. Internal initiatives include the vertical slice reviews, various configuration verification initiatives, and a series of Focused Design Reviews. In addition, audits, assessments and other oversight activities have been conducted. As these programs, reviews and initiatives have identified deficiencies, corrective actions have been or are in the process of being implemented.

Collectively, these activities, processes, testing programs, inspections, and initiatives provide reasonable assurance that SSC configuration and performance at BVPS are consistent with the facility design bases.

Section 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."

Section D discusses the problem identification and corrective action program at BVPS and its evolution since the licensing of BVPS 1 in 1976. It explains that for many years the program focused on safety significant issues that included concerns related to

consistency with the design bases. However, because of the importance of the ability to identify lower level issues, changes were made to the program in 1993. These changes were initiated, first, to reduce the threshold level for reporting problems, thereby broadening the program's focus to include the reporting of a wider range of conditions and issues with significance levels lower than had previously been reported, and, thereafter, to improve the ability to accurately disposition the increased number of issues reported. After implementation of these changes, there was a significant increase in the number of problem reports generated under the program, from approximately 200 reports in 1992 to over 1,800 reports in 1996. This increase caused an overload on the resources available to adequately resolve the issues identified in a timely manner. Once the impact of the increased number of issues was recognized, BVPS management implemented measures to accommodate the increase.

Section D discusses a Quality Assurance audit of the corrective action program performed by the Quality Services Unit in October 1996, and an independent assessment of the program by FPI International in November 1996, each of which, because of their retrospective nature, reiterated the need to modify the program to accommodate the increased number of problem reports. The Section also describes significant modifications made to the BVPS corrective action program, effective January 1, 1997, designed to address the concerns and to provide added assurance that conditions adverse to quality are identified, corrective actions to prevent recurrence are implemented and reporting to NRC is performed as required. Finally, Section D describes other problem identification and corrective action programs at BVPS such as (1) the Employee Concerns Resolution Program, which provides an alternative, confidential and impartial channel for communicating employee, contractor and consultant concerns for investigation and resolution through the Ombudsman position, and (2) the SSFE vertical slice review program, which incorporates the basic elements of a comprehensive corrective action program by identifying issues, prioritizing them in accordance with safety significance, establishing completion schedules and tracking issues to completion.

Section 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."

Section E sets forth DLC's reasons for concluding that its current programs and processes are effective overall in providing reasonable assurance that the configuration of BVPS is maintained consistent with its design bases. The Section looks again at the salient features of each of (1) BVPS's comprehensive and controlled engineering design and configuration control processes; (2) its processes for translating design bases requirements into appropriate operating, maintenance and testing procedures; (3) the policies, programs and procedures implemented at BVPS to maintain the configuration and performance of SSCs consistent with design bases; and (4) the recently re-engineered corrective action program. This Section also discusses the evaluation of three (3) recent BVPS events for their contributions to the analysis and conclusions of this Response, as well as the bases that provide reasonable assurance that the configuration of BVPS 1 and BVPS 2 is consistent with the design bases.

DLC's engineering design and configuration control processes and the outputs of these processes have been reviewed extensively, both internally and externally. Assessment results generally demonstrate that design and configuration control processes have been effective. As expected with any effective Quality Assurance program, deficiencies have been identified. Deficiencies were prioritized and corrective actions were dispositioned. These deficiencies have not called into question the reasonable assurance that BVPS 1 and BVPS 2 are configured and operated consistent with their design bases.

The effectiveness of the translation of design bases information into operating, maintenance and testing procedures has been assessed both internally and externally. Of special note are the results of the vertical slice reviews. Of the deviations that were found in the translation of design bases information into procedures, only a few were significant enough to be findings. Corrective actions have been or are in the process of being implemented.

The effectiveness of the configuration management process in maintaining consistency between the SSCs and their design bases has been assessed both internally and externally. Of special note again are the vertical slice reviews. As discussed in Section C, the SSCs were found to be consistent with their design bases. Of the differences between SSCs as configured and their design bases, only a few were significant enough to be findings. Corrective actions have been or are in the process of being implemented for the differences discovered. Additional vertical slice-type assessments will continue to be performed as a method to provide additional assurance that BVPS SSCs are consistent with the design bases. These circumstances support the conclusion that there is reasonable assurance that the configuration and performance of the BVPS 1 and BVPS 2 SSCs are consistent with their design bases.

Although problems were being reported and corrected under the prior corrective action program, and although some strengths in the previous program were noted by the NRC, DLC recently implemented significant enhancements to the corrective action program in response to management recognition of deficiencies with the then existing Problem Reporting program. These enhancements are designed to provide added assurance that conditions adverse to quality are identified, corrective actions to prevent recurrence are performed, and reporting to the NRC is performed as required.

Section E concludes that the programs and processes in place at BVPS, combined with DLC management's clear communication of expectations that the plant configuration be consistent with the design bases, implementation of a continuing critical self-checking process, results of assessments conducted to date and analysis of recent plant events discussed in this Response, provide reasonable assurance that the configuration of BVPS 1 and BVPS 2 is consistent with the design bases.

Section F. "In responding to items (a) through (e), indicate whether any design review or reconstitution programs have been undertaken, and if not, provide a rationale for not implementing such a program."

Examples of significant design review or reconstitution programs which have been undertaken by BVPS are discussed throughout this Response and again enumerated in Section F. These efforts included the ten (10) vertical slice reviews, two (2) focused Vertical Slice Reviews, development of Design Basis Documents (DBD), Design Basis Reconstitutions, Design Drawing Reconciliations, Electrical Calculation Upgrades, Seismic Qualifications, Environmental Qualification, Fire Protection and Containment Penetrations.

Section F also sets forth DLC's commitments as follows:

(1) To conduct at least one (1) vertical slice-type assessment at each Unit every other year, starting in 1997 at BVPS 2 and in 1998 at BVPS 1,

(2) To complete a detailed review of the BVPS 1 and BVPS 2 UFSARs by December 31, 1998, and

(3) To conduct a follow-up effectiveness review of the new Condition Reporting Program in 1997.

INTRODUCTION

PURPOSE

This document provides information in response to the NRC's request dated October 9, 1996, pursuant to 10 CFR 50.54(f), regarding the adequacy and availability of design bases information for BVPS. DLC understands the importance of designing, modifying, operating, configuring, and maintaining BVPS in accordance with its design bases. The following information and details regarding BVPS are submitted in response to questions posed by the NRC in its request, and are intended to provide added confidence that BVPS is operated and maintained within the design bases, and that any deviations from the design bases are reconciled in a timely manner, in accordance with the DLC Operations Quality Assurance Program.

In its request, the NRC specifically requested licensees to provide the following information:

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

The NRC further requested that, in responding to items (a) through (e), licensees indicate whether any design review or reconstitution programs have been undertaken, and if not, that licensees provide a rationale for not implementing such programs.

The following pages of this document provide the requested information relative to BVPS in the order presented above.

The details herein contain a description of the processes currently in use for engineering design and configuration control, as well as those processes for problem identification and disposition. These are evolutionary due to continuing assessment activities and the need to improve in recognition of ever increasing standards.

BACKGROUND

BVPS 1 and BVPS 2 are located on the same site in Shippingport Borough on the Ohio River in Beaver County, Pennsylvania, and are operated and maintained by the DLC Nuclear Power Division. Each Unit uses a pressurized water reactor nuclear steam supply system (NSSS) and turbine generator, both supplied by Westinghouse Electric Corporation. The balance of each Unit, including the containment structure, was designed and constructed by DLC and other co-owners, with assistance from the architectural engineering firm of Stone & Webster Engineering Corporation. Each NSSS is a three-loop design with a licensed reactor core power level of 2,652 MWt.

BVPS 1 received its construction permit in 1970. The BVPS 1 Final Safety Analysis Report (FSAR) follows the guidance of the Atomic Energy Commission publication, "A Guide for the Organization and Contents of Safety Analysis Reports" dated June 30, 1966. BVPS 1 received its operating license on January 30, 1976. BVPS 2 received its construction permit in 1974. The BVPS 2 FSAR was prepared following the guidance contained in Regulatory Guide 1.70, Revision 3, dated November 1978. BVPS 2 received its operating license on May 28, 1987. The eleven-year time span between the licensing of the two Units resulted in their being licensed under different regulatory requirements.

BVPS 1 experienced significant changes during the first ten years of operation. Reviews being conducted for a plant change resulted in the shutdown of the Unit and evaluation of the seismic adequacy of selected piping systems. This was a precursor to the NRC issuance of Inspection and Enforcement Bulletin No. 79-14. Additional changes occurred as a result of the NRC-issued TMI Action Plan. When BVPS 2 was licensed to operate in 1987, these industry events were reviewed as part of the initial licensing process. When the industry began organizing design document reconstitution programs, it was recognized that BVPS 1 could benefit through confirmation or the reconstitution of design bases information based on the industry processes in development at the time. A decision was made in 1988 to initiate vertical slice reviews, initially called SSFEs, of the design and operation of selected safety systems at BVPS 1. This effort evaluated the adequacy of the engineering design and configuration control processes and provided opportunities to reconstitute design bases information when necessary. This activity is discussed throughout this Response as one of the key methods for accounting for operating and maintaining BVPS 1 in accordance with its design bases as an earlier vintage plant.

As an owner and operator of the Units, DLC has instituted a program for quality assurance in nuclear power plant operations. It is described in a series of documents which provide an appropriate level of detail for the organizations to which those documents apply. At the highest level, DLC has implemented an approved Operations Quality Assurance Program, which meets the requirements of 10 CFR 50, Appendix B, defines the program requirements, specifies the responsibilities for implementing the program, and is applied to safety-related SSCs. Management expectations and general

direction for the operation and maintenance of BVPS are described in directives set forth in the Nuclear Power Division Administrative Manual. More detailed Nuclear Power Division Administrative Procedures describe the processes designed to meet those management expectations and establish formal interfaces among the various site departments. The various site departments are responsible for developing and implementing department-specific procedures to address these processes, including the interfaces with other departments. In this manner, senior management's expectations are translated into specific procedures and processes that enable individual departments to perform their specific responsibilities in support of the operation and maintenance of BVPS. The processes for engineering design and configuration control, as well as the processes for identifying problems and implementing corrective actions, are common to both BVPS 1 and BVPS 2. However, each Unit is operated by Unit-specific operating procedures, which incorporate the unique attributes of its individual license and Technical Specifications.

This Response describes the procedures and processes which are common to both BVPS 1 and BVPS 2. In addition, where reference to one Unit or the other is appropriate, Unit-specific details are provided. The differences between the two Units arose, in part, from the eleven-year time span between the issuance of their respective operating licenses. The different ages of the two Units resulted in the application of different criteria related to the licensing process applied by the NRC. These differences are expanded upon where appropriate within this Response.

RESPONSE PROCESS

To prepare this Response, DLC management formed a multi-disciplined team of knowledgeable and experienced BVPS employees to formulate a response plan, collect supporting information, organize information in support of the requested information, assess the adequacy of the gathered information, and prepare the submittal to be responsive to the NRC's request. A team charter was developed and contained a mission statement, goals and objectives, team membership, and a methodology for addressing the 10 CFR 50.54(f) questions.

The Response team collected descriptions of programs, processes, and assessments of elements related to: engineering design and configuration control; operating, maintenance and testing procedure development; SSC configuration and performance monitoring; problem identification and implementation of corrective actions, including actions to determine the extent of problems, action to prevent recurrence, and reporting to the NRC. In order to assess the implementation and effectiveness of these activities, relevant information was collected, including past self-initiated vertical slice reviews, departmental self-assessments, Quality Assurance audits, surveillance and assessment results, NRC SALP and inspection report results, and independent third party assessments.

The Response Team worked closely with BVPS management in preparing this Response. Management participation and review provided additional assurance that the information contained herein is accurate and responsive to the NRC's request for information, and that this submittal represents the collective knowledge of those responsible for operating and maintaining BVPS.

Separate Appendices have been developed to assist the reader in understanding the definitions and acronyms used herein. These Definitions are from various sources including 10 CFR 50.2, Operations QA Program, Administrative Procedures, Engineering Standards, NUREG-1397, NUMARC 90-12 and the NRC 50.54(f) letter dated October 9, 1996. See Appendix A for definitions and Appendix B for acronyms.

A. RESPONSE TO INFORMATION REQUEST (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

A.1 INTRODUCTION

This Section of DLC's Response overviews the engineering design and configuration control aspects of the Operations Quality Assurance Program which, among other things, guides the treatment of design bases issues at BVPS. It first focuses on the mechanisms used at BVPS to document design bases information, including a description of Design Basis Documents and Design Basis Reconstitutions. For purposes of this Response, "design bases information" encompasses design input documents, design studies or analyses, and design output documents that specify the design of an SSC. These are the reference documents which demonstrate that SSCs have been designed to perform their intended function within the reference bounds of the controlling parameters, and which provide a point of reference for future plant modifications.

Second, mechanisms are discussed that are designed to ensure continuing consistency with the design bases when facility modifications are made. These modification types include Design Changes, Design Equivalent Changes, Temporary Modifications, Administrative Changes, and Other Site Changes. Various processes, including Design Change Packages, Technical Evaluation Reports, Temporary Modifications, Administrative Changes and Other Site Changes control each of these types of modifications. These processes are described below.

Third, this Section addresses the configuration control process at BVPS; that is, the manner in which changes to the site are made under the umbrella of "configuration control." The configuration control process includes the work control, system configuration control, equipment control, and procedure control processes. (Of these four processes, all except procedure control are addressed in greater detail in Section C; the procedure control process is addressed in greater detail in Section B.) These core processes apply to, but are not limited to, new designs and revisions or changes in design. Other significant processes described in this Section include those used to implement 10 CFR 50.59 requirements, 10 CFR 50.71(e) requirements, license amendments, other design and configuration controls and programmatic improvements.

Finally, the vertical slice review program that has been used to provide additional verification of the design bases and plant as-built configuration is described. It has been considered a principal method for identifying and evaluating design bases issues.

The processes described above implement the engineering design and configuration control aspects of the Operations Quality Assurance Program at BVPS. This program

provides the foundation for the programs, procedures and processes used at BVPS. The Quality Assurance Program, developed and implemented prior to licensing of BVPS 1, was based on the eighteen (18) criteria of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The program describes the requirements and organizational responsibilities which govern the conduct of BVPS activities that affect the safety-related functions of SSCs. This includes the processes for design, configuration control and record maintenance that satisfy Criteria III (Design Control), V (Instructions, Procedures and Drawings) and XVII (Quality Assurance Records) of 10 CFR 50, Appendix B. Over time, other programs were established in response to DLC management initiatives, industry initiatives, NRC regulatory issues or the identification of deficiencies.

The specific provisions of the Operations Quality Assurance Program regarding the required control and maintenance of the design bases are implemented through a hierarchy of administrative documents. Several multi-tiered types of administrative documents are used to implement the broad policy objectives in the higher level documents. Among these are: (1) site administrative procedures, which establish the general administrative requirements and program details that apply to work activities on SSCs at BVPS and address the interface between departments; (2) department administrative procedures, which are more detailed than the site administrative procedures and which describe the detailed processes used in each department; and (3) Engineering Standards, the Plant Installation Process Standards, and work procedures (e.g. operating, maintenance and testing procedures). The site and department administrative procedures describe the processes used to implement and maintain engineering design and configuration control. Work procedures implement the requirements of administrative procedures and activities, including those described in the Technical Specifications, and Regulatory Guide 1.33, Rev. 2, February 1978, Appendix A, "Quality Assurance Program Requirements."

In terms of organization, the Nuclear Engineering Department is responsible and accountable for the overall plant design. Other organizations, both internal and external, may be delegated specific responsibilities provided the organization has a documented design program which meets ANSI N45.2.11, "Quality Assurance Requirements for Design of Nuclear Power Plants." The majority of engineering design at BVPS is performed by in-house engineering personnel. In certain instances, the DLC Nuclear Engineering Department also relies upon temporary engineering staff.

The Nuclear Engineering Administrative Procedures provide requirements for control of the design process. Engineering Standards provide additional administrative and technical guidance. The Plant Installation Process Standards provide generic installation standards and guidelines for specific installation activities, such as seismic conduit and supports.

Collectively, these procedures require that engineering design and configuration control activities be conducted and documented in accordance with the Operations Quality

Assurance Program. In particular, the following engineering design and configuration control processes address the information requested by the NRC:

- Design Bases Information Control
- Modification Control
- Configuration Control Process
- Training Related to Design and Configuration Control
- Implementation of 10 CFR 50.59
- Implementation of 10 CFR 50.71(e)
- License Amendment Requests
- Other Design and Configuration Controls
- Programmatic Improvements

Experience with the implementation of these processes provides additional assurance that the applicable design requirements such as design bases, regulations, codes and standards are correctly translated into applicable design documents, such as calculations, design analyses, specifications and drawings.

A.2 DESIGN BASES INFORMATION CONTROL

Overview

Engineering design and configuration control processes involve the evaluation and revision of design bases information. At BVPS, design bases information is included in design analyses, calculations, specifications, drawings, technical reports, Design Change Packages, licensing correspondence, UFSAR, Technical Specifications, procedures and other program documents. The majority of design bases information is available and controlled through the Beaver Valley Records Center. Design bases information may be reconstituted in accordance with established procedures when it is needed and not available.

Design analyses, specifically calculations, originate primarily from three (3) sources: DLC (including third-party calculations reviewed and approved by DLC), the Architect Engineer (Stone & Webster Engineering Corporation), and the NSSS vendor (Westinghouse Electric Corporation). These design analyses were developed by DLC, the Architect Engineer and the NSSS vendor under their respective approved Quality Assurance Programs. Some calculations, and other design bases information, are maintained by the Architect Engineer and the NSSS vendor in accordance with their approved Quality Assurance Programs. These are typically those of a proprietary nature to which DLC personnel have access as needed.

Independent design verifications are required for calculations associated with safety-related SSCs and with the seismic portions of non-safety-related SSCs to ensure that

appropriate design bases information has been reviewed and incorporated. These design verifications are conducted using either an alternate calculation method or by reviewing the appropriateness of inputs, assumptions, references, and applicable codes, standards, and regulations in accordance with ANSI N45.2.11. Calculations that are generated, reviewed and approved by DLC are maintained as DLC Quality Assurance records.

Procurement specifications transfer design requirements for components into procurement documents. Multi-discipline (Environmental Qualification, Seismic, Materials, Fire Protection, ASME) reviews of specifications are performed as appropriate. Procedures specify the criteria for determining when specialized expertise and review is needed. Following management approval, the specifications are forwarded to the Beaver Valley Records Center for retention.

Design drawings are used to document the design and as-built configurations of BVPS 1 and BVPS 2 and to maintain a record of installed modifications. The following are typical of the types of drawings which are maintained:

- Electrical: One-line Schematics, Elementary Circuit Diagrams, Wiring Diagrams, Logic Diagrams, Conduit and Conduit Support Diagrams
- Mechanical: Flow Diagrams, Piping Diagrams, Valve Operating Number Diagrams, In-Service Inspection Diagrams and Isometric Drawings
- Structural: Architectural Arrangement Diagrams, Concrete Drawings, Steel Structure Drawings, Facilities Drawings, and the Site Map
- Various vendor supplied drawings

Design drawings are typically created or revised as the result of a modification. Prior to issuance, drawings receive multi-level review. Depending on the complexity and scope of the change, they may also receive an independent review which is intended to ensure that necessary design information has been incorporated. Design drawings are stored in the Beaver Valley Records Center and are distributed and controlled in accordance with the BVPS Document Control Program.

The Normal System Arrangement Valve List(s), contained in the Operating Manuals, and the Valve Operating Number Diagrams are the controlling documents that reflect the normal operating configuration of each BVPS system. The current operating configuration is indicated by the annotated Valve Operating Number Diagrams, which are maintained in the Control Room.

Procedures for design bases information programs contain provisions for the generation, maintenance, revision and control of design information including design analyses, calculations, specifications and drawings. These procedures are designed to ensure that documents which describe or control activities affecting the operation of BVPS are reviewed for adequacy, approved for release by authorized personnel, and issued in a controlled manner. The document control and records management programs establish requirements for the transmittal, distribution, indexing, storage, retention and

retrievability of design information. Collectively, these processes provide control of design bases information and assure compliance with 10 CFR 50, Appendix B.

A.2.1 Design Basis Documents

One way that design bases information at BVPS 1 and BVPS 2 can be referenced is through the Design Basis Documents. A Design Basis Document Program was initiated at BVPS in 1988 in an effort to organize and collate the design bases information of the then recently licensed BVPS 2. This program was later expanded to include BVPS 1 and was continued under the industry initiative outlined in NUMARC 90-12. A total of 111 Design Basis Documents (89 system-related and 22 generic) address the majority of BVPS safety-related systems. In general, the Design Basis Documents include functional requirements, performance parameters and general design requirements, such as applicable codes, standards, environmental qualification, missile protection and material considerations. The Appendices to the Design Basis Documents include references to major equipment, calculations, drawings and Design Change Packages. The Design Basis Documents are intended to aid in understanding and accessing design bases information at BVPS. These Design Basis Documents are used as references to identify design bases information that is available and controlled through the Beaver Valley Records Center.

Design Basis Document changes are processed using the Design Basis Document Revision Request Form. These requests are required to be submitted when changes are made to the design bases information referenced in the Design Basis Document (such as reconstituted calculations). A multi-tiered priority system is established for Design Basis Document control and issuance that considers such things as the safety significance and Maintenance Rule impact of the system.

A.2.2 Design Basis Reconstitution

If needed design bases information is unavailable, it must be reconstituted. The Design Basis Reconstitution procedure establishes requirements for developing, verifying and validating safety-related design bases information. This procedure is designed to ensure consistency between the Design Basis Documents, design documents and the current Unit configuration. The procedure also provides for the reconstitution or use of alternate methods when needed design calculations or analyses are missing or inadequate. Discrepancies noted between design documents and the as-built BVPS SSC configuration are required to be reported in accordance with the Condition Report system (described in Section D) for resolution or corrective action.

A.3 MODIFICATION CONTROL

A.3.1 Introduction

BVPS has adopted a graded modification control program that applies to five (5) types of modifications. They are:

- Design Changes,
- Design Equivalent Changes,
- Temporary Modifications,
- Administrative Changes, and
- Other Site Changes.

The BVPS Modification Control Program defines the responsibilities of each site department in the modification control processes. The overall program, which meets the requirements of 10 CFR 50, Appendix B and ANSI N45.2.11, controls:

- the selection of design inputs,
- the translation of design inputs into design outputs, including incorporation into design documents such as drawings, specifications, and procedures,
- the preparation, independent review and approval of process and design output documents, and
- the verification of design outputs.

BVPS modification control processes are described in site administrative procedures. Implementation of these controls is intended to ensure that modifications are consistent with the design bases.

Each type of modification is discussed below with regard to maintenance of consistency with the design bases. Modifications characterized as Design Changes include Design Change Packages, Software Changes, Setpoint Changes, and Equipment Removal/Retirement In-Place Evaluations. Design Equivalent Changes include Design Equivalent Evaluations, Vendor Technical Information Updates, Design Document Discrepancy Resolutions, and Communication System Changes. Administrative Changes consist of UFSAR updates, Technical Specification changes and procedure changes. As discussed below, different processes apply depending on the modification type.

A.3.2 Design Changes

Design Changes are defined as modifications that change design requirements governing performance of SSCs' design bases. The Design Change Package process is used to control and document design changes.

Other types of Design Changes include Software Changes, Setpoint Changes and Equipment Removal/Retirement In-Place Evaluations. These types of Design Changes are not controlled by the Design Change Package process. Rather, Software Changes are administratively controlled by the Software Configuration Management Program (described below in Section A.3.2.2). Setpoint Changes and Equipment Removal/Retirement In-Place Evaluations are administratively controlled by either the

Technical Evaluation Report process or other department specific methods (described below in Section A.3.2.3).

A.3.2.1 Design Change Package Process

A.3.2.1.1 Modification Requests

The Design Change Package process begins with a modification request. It can be initiated by any individual at BVPS. It is communicated by Engineering Memorandum to the System and Performance Engineering Department. When the System and Performance Engineering Department determines that the proposed modification constitutes a necessary Design Change, a Design Change Request is generated and presented to the Preliminary Review Committee for review and approval. The Preliminary Review Committee consists of management personnel from various site departments. When a Design Change Request has been approved by the Preliminary Review Committee a Design Change Package is initiated.

A.3.2.1.2 Design Input

The Nuclear Engineering Department is responsible for the overall engineering of the Design Change Package. At the beginning of the design phase of the Design Change Package, the Engineering Review Screen is used to determine applicability of design bases requirements such as: equipment qualification, seismic qualification, fire protection, modifications to concrete walls, heat loads and ventilation, missile generation, digital or programmable equipment, electrical loading, simulator impact, station blackout equipment, NRC Generic Letter 89-10 Motor Operated Valve (MOV) management, ASME Section XI, Maintenance Rule and Technical Specifications. In addition, this screen addresses single failure and redundancy requirements, and the effects on Emergency Operating Procedures (EOPs) and Valve Operating Number Diagrams.

A Design Input List, based on ANSI N45.2.11, is prepared for safety-related, seismic and station blackout-related Design Change Packages. This list considers design bases information, codes, standards and NRC requirements. Sources for design inputs, along with design parameters, limitations, and conditions are identified. During the Design Change Package process, changes to the Design Input List, including the reasons for the changes, are required to be identified, approved, documented and controlled. Design Input List development is thus a means of providing that design bases information is retrieved and used in the Design Change Package process.

A.3.2.1.3 Package Development

During the Design Change Package process, other applicable activities are performed, including the development of an Installation and Test Package, conduct of design reviews, the assignment of Equipment Identification Numbers for new equipment, and the development of procurement specifications, spare parts lists and setpoint changes for affected equipment.

The Nuclear Engineering Department prepares the Installation and Test Package for the Design Change which is reviewed by other site departments. Design bases requirements are factored into the development of the Installation and Test Package, which contains summary information, installation and testing requirements, detailed installation instructions and marked-up drawings. The Installation and Test Package provides the sequence for installation, testing, and operational acceptance of the Design Change Package. It also identifies proof, functional, and operational tests required by applicable codes and licensing commitments, and the parameters required to develop testing procedures.

A.3.2.1.4 Review and Approval

Various site departments participate in a Design Review meeting prior to the issuance of the Installation and Test Package. The purpose of these meetings is to review the objectives, approach, scope, operability, maintainability, testability, and schedule, and to elicit end-user feedback.

For each Design Change Package, a 10 CFR 50.59 screen is completed and a 10 CFR 50.59 safety evaluation is performed as required. If it is determined that an unreviewed safety question exists, or that a change to the Technical Specification is required, NRC approval is obtained prior to installation of the modification. Additional information regarding the 10 CFR 50.59 process is contained in Section A.6 of this Response.

The responsible engineer(s) and Nuclear Engineering Department management perform a multi-discipline review of the Design Change Package. An independent review is then conducted to verify the accuracy and completeness of the Design Change Package documentation and that open issues from the Design Review meetings have been addressed. In addition, an independent design verification, utilizing a Design Review Checklist based on ANSI N45.2.11, is completed for safety-related and seismic Design Change Packages. This verification must be completed prior to operational acceptance.

In addition to these reviews, Design Change Packages are also reviewed by the Onsite Safety Committee (OSC)[1]. This review is intended to ensure that design bases information has been properly considered. Refer to the discussion on 10 CFR 50.59 in Section A.6 below for additional details of OSC review of Design Change Packages.

Following review and management approval, the Design Change Package is transmitted to the Beaver Valley Records Center. The Beaver Valley Records Center provides controlled distribution of the appropriate Design Change Package documentation to the implementing departments.

---

[1] The OSC includes a Chairman and nine (9) other members drawn from various organizations and disciplines as identified in the Technical Specifications.

The responsible engineer notifies other site departments, such as Operations and Maintenance Procedures, Training, Simulator and Chemistry, of the impending Design Change Package installation. These advance notifications enable the affected departments to update their respective programs to support the operational acceptance of the Design Change Package. Additional details regarding the translation of design information changes into affected programs and procedures are provided in Sections B and C of this Response.

A.3.2.1.5 Installation and Revisions

The Installation and Test Package is issued to the installation organization for installation and testing. Installation, including Quality Control inspections and hold points, is performed in accordance with the Installation and Test Package. Technical revisions to the Installation and Test Package that are identified during the installation and testing phases are reviewed, approved and documented on Engineering Change Notices or Field Change Notices. These revisions are also evaluated to determine their effect, if any, on the scope of the modification. Changes in scope are those which alter the applicable SSC's design, function, or method of performing that were previously reviewed by the OSC. If a change in scope results, the applicable 10 CFR 50.59 safety evaluation is revised and resubmitted for OSC review prior to the re-issuance of the change document(s) and the implementation of the change.

A.3.2.1.6 Modification Testing

Design Change Package modification testing includes proof, functional and operational testing. Proof testing is performed as necessary to verify that the Design Change Package installation was accomplished in a satisfactory manner and that it meets the design intent (e.g., component hydrostatic test, leak test, control circuit test). Functional testing is performed to verify that the systems and equipment meet their intended function within the specified limits (e.g., closing time of valves, pump flow rates). Operational testing is performed to verify proper overall system performance in conjunction with other systems. The modification is accepted based on satisfactory performance of required testing and acceptable resolution of any quality-related open or nonconforming issues. Test results are documented and evaluated to the established acceptance criteria and testing requirements. The satisfactory completion of testing provides additional verification that design requirements have been met. The surveillance testing that is required following installation is addressed in Section B of this Response.

A.3.2.1.7 Operational Acceptance

Following installation and testing, the responsible engineer completes a Turnover Checklist prior to operational acceptance which verifies that appropriate design output documents have been updated. Nuclear Engineering, Operations and Maintenance Department personnel jointly perform a pre-turnover walkdown to verify that the design was correctly installed. Before the SSC can be placed into service and declared operable,

the General Manager, Nuclear Operations or his designee must approve the Design Change Package operational acceptance documentation.

A.3.2.1.8 Records Update

The Nuclear Engineering Department is responsible for ensuring design documents affected by the Design Change are revised within the time constraints specified in site procedures. Records update is also discussed in Section A.9.3, below.

A.3.2.2 Software Change Process

Design Changes to software are covered by the Software Change process. The Software Configuration Management Program requires that applicable computer software be developed, purchased, used and maintained in accordance with 10 CFR 50, Appendix B requirements. This program applies to application software for programmable controllers and plant process computer systems.

Changes to programmable controllers and plant process computer systems are evaluated for applicability and impact on plant systems. These evaluation results are reviewed against design documents, such as drawings and specifications. If the change is acceptable, a requirement specification is generated and an implementing software design package written. Following design package approval, the software code is modified. The software change process undergoes an independent verification and validation process to determine whether the software module performs correctly, and that unintended functions have not been introduced.

A.3.2.3 Other Design Change Processes

Other specific types of design changes are controlled by specifically tailored change processes. The Technical Evaluation Report process is used to control and document engineering evaluations in accordance with the Operations Quality Assurance Program. Examples of evaluations controlled by Technical Evaluation Reports that are design changes include selected Setpoint Changes and Equipment Removal/Retirement In-Place Evaluations. The Technical Evaluation Report process is discussed in further detail in Section A.3.3.1, below.

The Nuclear Engineering Department prepares Technical Evaluation Reports to document and control process setpoint changes for safety and Technical Specification-related equipment, except for those relating to radiation monitors which are discussed below. The setpoint control process addresses NRC Regulatory Guide 1.105 "Instrument Setpoints." Setpoint changes are screened for 10 CFR 50.59 applicability. If required, a 10 CFR 50.59 safety evaluation is prepared and independently reviewed. Setpoint Change Technical Evaluation Reports are reviewed by the OSC and approved by the General Manager, Nuclear Operations or his designee.

The authorization, implementation, and tracking of setpoint and database changes associated with the BVPS Radiation Monitoring Systems are performed according to Health Physics Department procedures or other existing Health Physics controls. The Health Physics procedures address the authorization, implementation, tracking, independent verification and required notifications. The setpoint/database change forms are forwarded to the Beaver Valley Records Center for records retention.

A.3.3 Design Equivalent Changes

The second type of modification is the Design Equivalent Change. A Design Equivalent Change is defined as a modification to an SSC that does not change design requirements governing performance of the design bases for that SSC. These include modifications to an SSC that will not: (1) alter its ability to meet its design bases, normal or accident operating requirements; (2) have a detrimental effect on the operation of the system where it is used; or (3) require extensive modification to allow it to fit. The Technical Evaluation Report process is used to control and document Design Equivalent Changes. Design Equivalent Changes administered by Technical Evaluation Reports include Vendor Technical Information changes, Design Document Discrepancies, and Communication System Changes.

A.3.3.1 Technical Evaluation Report Process

The Technical Evaluation Report package includes summary sheets, evaluation sheets with the appropriate marked-up attachments, a programs impact checklist[2], the conclusion and basis for the conclusion. Also included if appropriate are any special installation and testing instructions. Depending upon the type of evaluation, the process requires that an Impact to Programs Checklist be completed to evaluate design bases concerns. The Impact to Programs Checklist considers the programmatic effects (e.g., environmental qualifications, seismic, electric loads, etc.) and is similar to that which is performed in the Design Change Package Engineering Review Screen.

The responsible engineer is required to complete a 10 CFR 50.59 Screening Form[3] for Technical Evaluation Reports. If applicable, a 10 CFR 50.59 safety evaluation is completed. The 10 CFR 50.59 process is discussed further in Section A.6, below.

Multi-disciplined reviews of the Technical Evaluation Report are performed based upon the results of the Impact to Programs Checklist. Independent technical reviews are completed based on the complexity and scope of the Technical Evaluation Report. A supervisory review of the Technical Evaluation Report is performed and is intended to ensure its technical adequacy and administrative completeness. The OSC reviews Design

---

[2] Program impact checklists are not required for Vendor Technical Information and Design Document Discrepancy type evaluations.

[3] 10 CFR 50.59 Screening Forms are not required for Vendor Technical Information and Design Document Discrepancy type evaluations.

Equivalent Change Technical Evaluation Reports involving safety-related SSCs and other Technical Evaluation Report types based upon criteria described in the process procedures. Technical Evaluation Reports related to Setpoint Changes or Equipment Removal/Retirement In-Place Changes are also reviewed and approved by the General Manager, Nuclear Operations or his designee. Following approval, the Technical Evaluation Report is issued through the Beaver Valley Records Center.

The engineer responsible for the Technical Evaluation Report develops special installation and test requirements that are necessary to enable detailed work instructions to be developed. The Maintenance Program Unit uses Maintenance Work Requests to translate these requirements into the necessary work instructions for ensuring the proper installation and control of modifications which are approved by the Technical Evaluation Report.

Following installation and testing of a Technical Evaluation Report modification, the Nuclear Shift Supervisor reviews the operational acceptance documentation before placing the SSC into service. Design documents affected by the Technical Evaluation Report are updated accordingly. Technical Evaluation Reports follow records updating requirements similar to that of Design Change Packages.

A.3.3.2 Other Technical Evaluation Reports

In addition to the modification types discussed above, Technical Evaluation Reports are generated to document other engineering evaluations, such as reactor core modifications. DLC provides information to Westinghouse Electric Corporation (fuel vendor) regarding design or operating changes implemented or planned for the current fuel cycle. This information is used by Westinghouse in the design of the reactor core. Westinghouse transmits to DLC a Nuclear Design Report which contains design bases information for the next fuel cycle. Westinghouse performs a safety evaluation of the proposed reactor core modifications. The DLC Nuclear Engineering Department performs a 10 CFR 50.59 safety evaluation of the proposed reactor core modifications for submittal to the OSC. DLC incorporates design information from the Nuclear Design Report into applicable site operating procedures.

Technical Evaluation Reports are also generated to document evaluations for Quality Assurance Classification Determinations, or equipment additions that do not require an Equipment Identification Number (e.g., ladders, railings, grating, etc.).

A.3.4 Temporary Modifications

Temporary Modifications constitute a third kind of modification which is used to maintain a system or component operable for a limited duration. Temporary Modifications require an evaluation of technical attributes and a 10 CFR 50.59 safety evaluation prior to installation. An independent reviewer, appropriate site management personnel and the OSC review the Temporary Modification.

The General Manager, Nuclear Operations or his designee authorizes installation of safety-related Temporary Modifications. Installation of non-safety-related Temporary Modifications may be approved by the Nuclear Shift Supervisor or a higher management level. While the Temporary Modification is in effect, Nuclear Operations maintains the Temporary Modification documentation in a permanent logbook located in the Control Room. A copy of the Temporary Modification documentation is stored in the Beaver Valley Records Center.

Affected design drawings and Valve Operating Number Diagrams which are located in the Control Room are required to be annotated to reflect Temporary Modifications. Jumper and Lifted Lead Tags are installed on affected equipment, except when tags would be located in the reactor containment during power operation. Appropriate Nuclear Engineering Department personnel are required to review the Temporary Modification documentation to identify impacts on the overall plant design. Both Nuclear Engineering Department and System and Performance Engineering Department personnel are required to review the Temporary Modification documentation to maintain awareness of current system configuration.

Nuclear Operations reviews outstanding Temporary Modifications quarterly. The system is restored by removing Temporary Modifications within six months or the next Refueling Outage following installation. The General Manager, Nuclear Operations or his designee can extend a Temporary Modification with written justification. An independent verification is performed of the Temporary Modification removal and system restoration. The annotations are removed from the design drawings to reflect the restored configuration. Following Temporary Modification removal, the completed Temporary Modification documentation is forwarded to the Beaver Valley Records Center for retention. A Design Change Package or Technical Evaluation Report is necessary to incorporate a Temporary Modification into a permanent plant design.

A.3.5 Administrative Changes

The fourth type of modification, the Administrative Change, is controlled by specific site procedures. Administrative Changes include UFSAR updates, Technical Specification changes, Operations, Maintenance, Testing and Instrument and Control Procedure changes, and Operating Manual (including Normal System Arrangement) changes. The processes for UFSAR updates and Technical Specification changes are described in Sections A.7 and A.8, respectively. The processes used to make changes to the operating, maintenance, and testing procedures are described in Section B of this Response. Changes to the Normal System Arrangement are controlled by the Nuclear Operations Unit administrative procedures.

A 10 CFR 50.59 screening is required to be performed for changes to the Normal System Arrangement of an operating system that are not controlled by procedure or equipment clearance.

A.3.6 Other Site Changes

The fifth modification type is described as "Other Site Changes." These changes are defined as modifications to the site facility that do not involve design requirements that govern performance of SSCs. These modifications are implemented in accordance with established work control procedures, policies and standards and require 10 CFR 50.59 screening. A 10 CFR 50.59 safety evaluation is performed, if necessary.

A.4 CONFIGURATION CONTROL PROCESS

A.4.1 Introduction

The Configuration Control Program at BVPS is designed to assure that the as-built configuration of the Units reflects the current design bases. Configuration control is composed of an inter-related set of processes that include engineering design, modifications, configuration and design bases changes. The overall objectives of configuration control are:

- Verification of the design bases
- Documentation of design bases
- Continued maintenance and updating of the design bases as modifications and changes in criteria and requirements are effected
- Verification that modifications and changes in criteria and requirements are consistent with established requirements for the balance of the plant not affected by these changes

These objectives are met through the implementation of specific procedures that were established to meet these objectives and more generally by the Operations Quality Assurance Program. These procedures describe the processes for work control, system control, equipment control and procedure control necessary to maintain SSC configuration and performance consistent with the design bases. These processes are summarized below.

A.4.2 Work Control Process

Maintenance Work Requests are used to control and document maintenance work activities on SSCs at BVPS. Based on the Quality Assurance Category and work scope, appropriate Quality Control inspection requirements are identified. The affected Unit's Nuclear Shift Supervisor is required to approve the Maintenance Work Request prior to the performance of work. Following completion of maintenance work and prior to equipment being declared operable, post-maintenance testing is performed to ensure the SSC configuration and performance remain consistent with the design bases. Additional details regarding work process controls are provided in Section C of this Response.

A.4.3 System Configuration Controls

Operating procedures require system configurations to be consistent with Normal System Arrangement. Operations personnel perform routine tours to verify proper plant operation and to identify degrading trends and material condition. The BVPS Operating Manual provides instructions for implementing changes to the system Normal System Arrangement and verifying that the system is properly configured. Additional details regarding system configuration controls are provided in Section C of this Response.

A.4.4 Equipment Controls

The site's Tagging and Labeling Program standardizes the nomenclature, numbering, temporary labeling, color coding and tagging of SSCs. Equipment Identification Numbers are used for components identified for control under the BVPS Configuration Management Program.

The Master Equipment List is an on-line computer database information management system which is used to identify and maintain component configuration control. The Master Equipment List forms the nucleus of other site integrated databases to support configuration management and design bases conformance. At the component level, the Master Equipment List provides a link between engineering design attributes, design documents, spare parts and information regarding the installed configuration. The Master Equipment List is updated to reflect changes to component information resulting from a modification. Selected Master Equipment List database fields are required to be validated annually to ensure database accuracy.

In addition to the Master Equipment List, other controlled programs are used to document SSC configuration. These programs include the Environmental Qualification (10 CFR 50.49) Master List, the 10 CFR 50, Appendix R (BVPS 1) and Safe Shutdown Reports (BVPS 2), Nuclear Plant Reliability Data System, Setpoint Control Program and the Regulatory Guide 1.97 (Post Accident Monitoring) Instruments List. Additional details regarding the site's equipment controls are provided in Section C of this Response.

A.4.5 Procedure Control

Procedures are typically classified as administrative or technical. Administrative procedures describe the administrative requirements, responsibilities, activities, or actions needed to implement or comply with regulatory requirements and commitments. Technical procedures are used to document and describe the series of actions or steps necessary to accomplish a desired objective such as the performance of system operation, maintenance and testing.

The administrative procedure control process governs the development, review and approval of technical, work and implementing procedures. The process is integrated with

the safety evaluation process, described below, to apply the requirements of 10 CFR 50.59 for both new procedures and procedure revisions.

Multi-level reviews are performed to verify that procedures and procedure changes are administratively and technically accurate, consistent with regulatory requirements and commitments, and in accordance with the design and licensing bases. As required by the Technical Specifications, the OSC reviews new procedures and revisions that change the procedure intent. Following OSC review, the General Manager, Nuclear Operations or his designee approves the new procedure or procedure revision. Approved procedures and procedure changes are controlled in accordance with the BVPS Document Control Program.

As described in the Technical Specifications, site personnel can perform temporary changes to procedures provided that:

- The intent of the original procedure is not altered
- The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the affected Unit, and
- The change is documented, reviewed by the OSC and approved by the General Manager, Nuclear Operations or his designee within 14 days of implementation

Additional details regarding procedure controls are provided in Section B of this Response.

A.5 TRAINING RELATED TO DESIGN AND CONFIGURATION CONTROL

Training is conducted at BVPS to improve understanding of the design bases information, design control and configuration control. In accordance with 10 CFR 50.120, DLC maintains programs for the training and qualification of operations, maintenance, radiological protection, chemistry, and engineering personnel. These training programs are periodically evaluated against the Quality Assurance Program requirements as well as updated to reflect industry experience and changes to the facility, procedures and regulations.

Personnel who require unescorted station access receive General Employee Training and are required to attend General Employee Refresher Training annually. These training classes address employee responsibility to identify and document design and configuration-related conditions.

Licensed nuclear operations personnel receive training on the Technical Specifications, accident analysis, core damage mitigation and the EOPs. The Nuclear Shift Supervisors are trained on processes involving plant modifications, procedure and setpoint changes.

In addition, this training emphasizes the importance of considering design bases in the conduct of operations.

Department-specific training consisting of required reading and classroom instruction is provided for engineering support personnel. This training is designed to improve the overall knowledge and skill base of personnel to perform design engineering and configuration control-related activities. Training topics include new and revised programs (e.g., Maintenance Rule, Condition Reporting, modification processes), procedures, performance trends, and evaluations and inspections (e.g., Quality Assurance audits and NRC inspections).

A.6 IMPLEMENTATION OF 10 CFR 50.59

BVPS's process for implementing 10 CFR 50.59 serves as the means for determining whether proposed changes to the plant or procedures described in the UFSAR under the design and configuration control processes discussed above result in an unreviewed safety question. Activities characterized as changes, tests or experiments are subject to a 10 CFR 50.59 screening based on NSAC-125. The NRC's recent views on NSAC-125 guidance are being taken into account as described below. The 10 CFR 50.59 screening process is documented in accordance with the administrative procedures. This screening determines whether a 10 CFR 50.59 safety evaluation is required for the proposed activity. The OSC reviews documents that are screened for 10 CFR 50.59 applicability with the exception of non-intent procedural changes and Design Equivalent Change Technical Evaluation Reports as identified in department administrative procedures. The 10 CFR 50.59 process is applied to the following:

- Test or experiments not described in the UFSAR
- Changes that affect the existing design, function, or method of performing the function of a SSC as described in the UFSAR
- Changes that affect information relied upon for the operating license
- Temporary Modifications (bypasses, jumpers, lifted leads, etc.)
- Changes that affect the radiological waste system or equipment qualification
- Changes to procedures as described in the UFSAR
- Basis for Continued Operation that is to be permanently incorporated.

If the 10 CFR 50.59 safety evaluation identifies an unreviewed safety question, the Nuclear Safety Department is notified. The Nuclear Safety Department is responsible for preparing a license amendment request for NRC review and approval prior to implementation in accordance with 10 CFR 50.90.

The 10 CFR 50.59 safety evaluation documents the analysis of design and licensing bases issues and contains the following specific information:

- Description of and reason for the proposed change

- Applicable section of the UFSAR, Technical Specification acceptance limits and other references used to conduct the evaluation
- Unreviewed Safety Question Determination
- Reviews and approvals

Personnel are required to meet specific experience and training requirements to become qualified to prepare and review 10 CFR 50.59 safety evaluations. Initial 10 CFR 50.59 training includes a session that addresses the history, purpose and applicability of 10 CFR 50.59. The initial training also includes preparing a practice safety evaluation and identifying unreviewed safety questions. Recent NRC guidance was included in 1996 training by adding material on NRC Part 9900: 10 CFR Guidance, '10 CFR 50.59 Interim Guidance On The Requirements Related To Changes To Facilities, Procedures And Tests (Or Experiment)" and the current NRC positions regarding 10 CFR 50.59 evaluation of safety probabilities, safety margins and degraded operability conditions. Personnel are required to perform at least one (1) safety evaluation during the calendar year and to attend an annual retraining session in order to remain qualified.

The OSC reviews 10 CFR 50.59 safety evaluations generated at BVPS, along with their associated documentation. These items are distributed to the OSC members prior to scheduled meetings, or as the OSC Chairman determines, to allow sufficient time for a thorough review. The meetings provide a forum for design bases information to be discussed, since the committee is comprised of various departments, and the members are encouraged to adopt a questioning attitude with regard to the material presented. The OSC serves to independently assess the accuracy of this information by review of other controlled design documentation. Deviations noted as a result of this process are tracked as open items. The Safety Evaluation Subcommittee of the Offsite Review Committee (ORC) provides additional oversight by reviewing 10 CFR 50.59 safety evaluations.

A.7 IMPLEMENTATION OF 10 CFR 50.71(e)

The BVPS 1 and BVPS 2 UFSARs are currently updated annually in accordance with 10 CFR 50.71(e). The Nuclear Safety Department is responsible for the administration of the process for updating the UFSARs. Text portions of the UFSARs are available in a word searchable electronic database on a local area network. Changes to the UFSAR are normally the result of plant or procedure changes, identified inconsistencies between the UFSAR and the as-built configuration, or new or revised regulations. A 10 CFR 50.59 safety evaluation, reviewed by the OSC, supports changes to the UFSAR. A safety evaluation is not prepared when a change is editorial in nature. The process for updating the UFSARs consists of documenting the reason for the change and attaching a copy of the affected portion of the UFSAR which has been annotated to indicate the proposed change. The proposed UFSAR revisions and supporting information are submitted to departments selected by the preparer for review. Following resolution of comments, the UFSAR revision, including applicable department reviews, is forwarded to the Nuclear Safety Department for administrative review and approval. A log of requested changes to

the UFSAR is maintained and each change is tracked to a final disposition (i.e., implementation in a specific revision of the UFSAR). A recent change to this process will result in requested changes receiving several checks prior to final incorporation of the change into the UFSAR. These checks include:

- confirming the technical review is complete,
- confirming the change meets the NRC criteria for updating the UFSAR,
- confirming that the proposed changes conform to the format and content commitments of the existing UFSAR, and
- a determination that an adequate level of detail is provided.

Following this review, the UFSAR revisions are accumulated, incorporated, and forwarded to the Beaver Valley Records Center for controlled distribution to UFSAR copy holders. Distribution to the NRC is completed in accordance with regulatory requirements.

Also, as a result of the recent process change, the annual UFSAR update package will be assessed to monitor implementation of the update process. This assessment may be performed after the incorporation of the UFSAR changes. The suggested assessment topics are:

- consistency of the change with other affected UFSAR sections,
- adequate resolution of technical review comments, and
- confirmation that the UFSAR change does not exceed the scope of the supporting safety evaluation.

A.8 LICENSE AMENDMENT REQUEST

A license amendment request supports revising the operating license or the Technical Specifications and is one method of modifying the design bases. The BVPS 1 and BVPS 2 Technical Specifications are modeled after the Standard Technical Specifications for Westinghouse-designed pressurized water reactors (NUREG 0452). License amendment requests are processed in accordance with 10 CFR 50.90 and 10 CFR 50.91. The Nuclear Safety Department is responsible for the administration of the change control process. License amendment requests typically are a result of assessments of routine operating activities, NRC correspondence, and corrective actions for identified problems. Such requests are forwarded to or are initiated by the Nuclear Safety Department for the development and submittal of a license amendment request for NRC review and approval.

The preparation and submittal of a license amendment request consist of documenting the justification for the request, preparing a safety analysis describing why the proposed change is safe to implement, and preparing a statement of "no significant hazards" consideration. The license amendment request relies upon the information contained in

the UFSAR, design documents, and licensing documents. Analytical details and design bases assumptions which support the application are obtained or reviewed prior to presenting the proposed change to the OSC. Independent reviews are documented by those organizations affected by the proposed change. The Nuclear Safety Review Board (NSRB) and the ORC also review license amendment requests prior to submittal to the NRC.

The procedures affected by the proposed license amendment request are identified in parallel with the NRC review effort. Changes to procedures, programs or administrative controls required to implement a Technical Specification amendment are revised and reviewed for adequacy. Issuance of these procedures is coordinated with the amendment implementation date. Proposed procedure revisions which support Unit operations but are not currently required (e.g., required to support outage activities) are scheduled for implementation prior to the next scheduled surveillance activity.

A.9 OTHER DESIGN AND CONFIGURATION CONTROLS

A.9.1 Vertical Slice Reviews

In addition to the procedures and processes that are required under 10 CFR 50, Appendix B, DLC has established a program of vertical slice reviews for selected safety systems at BVPS. As implemented to date, these vertical slice reviews have consisted of eight (8) Safety System Functional Evaluations (SSFEs), a Service Water Operational Performance Inspection (SWOPI) and a Safety System Functional Inspection (SSFI) each modeled after NRC inspection guidance. These vertical slice reviews have focused on BVPS 1 because plants of its vintage were licensed to less stringent expectations than those applied to newer plants such as BVPS 2.

The SSFE Program was initiated in 1988 as an integrated approach to the identification, evaluation and reconciliation of the BVPS 1 licensing and design bases with the configuration baseline for safety-related systems in the Unit. The program subsequently evolved to include generic issues identified in several SSFEs. The SSFE Program is formally structured, conducted and managed. A program manual describes the overall program and individual responsibilities. Conduct of the SSFEs is detailed in a procedure modeled on the NRC's Safety System Functional Inspection methodology. Overall program direction and assessments of the safety significance of the issues identified were assigned to a Management Oversight Committee. The committee was also responsible for assuring the completion of corrective actions taken to address SSFE results.

In recognition of changing expectations for vertical slice reviews, the process was revised in 1992. Instead of continuing to apply the SSFE process, the two most recent reviews were conducted using specially developed and approved inspection programs and plans. These inspection programs and plans were based on the NRC's SWOPI and updated Safety System Functional Inspection methodologies, respectively. The plans also include detailed descriptions of the structure, conduct, management and implementation of the vertical slice reviews.

Systems were chosen for vertical slice reviews based on their safety significance and the site and industry-related issues at that time. Since 1988, vertical slice reviews have been conducted on ten (10) safety-related systems at BVPS 1 and one (1) safety-related system at BVPS 2 to verify that they are capable of performing their intended safety functions under normal operating and accident conditions.

The BVPS 1 systems were:

- Auxiliary Feedwater System (1988)
- Quench Spray System (1988)
- Emergency Diesel Generator (1988)
- Reactor Plant River Water System (1989)
- Recirculation Spray System (1989)
- Residual Heat Removal System (1989)
- Supplementary Leak Collection and Release System (1990)
- Electrical Distribution System (1991)
- River Water System (SWOPI) (1994)
- Safety Injection System (SSFI) (1995)

The BVPS 2 system was the Service Water System (SWOPI) (1994).

Each of the above reviews was conducted by a highly experienced, multi-disciplinary team that included representatives from external organizations. Detailed checklists were used. The results were extensively documented in formal reports.

In each case, the safety system evaluated was found to be capable of performing its intended safety function. In some cases, however, immediate actions were taken to improve safety system readiness. Each of the SSFEs identified a few findings, issues which had a level of safety significance that warranted priority attention, and several observations, issues of lesser safety significance, that could be addressed on a more deliberate schedule.

Details of these SSFE findings and observations, SWOPI plant issues and SSFI deficiencies as they relate to the consistency between the design bases and the procedures into which they are incorporated are provided in Section B of this Response. Details concerning the consistency between the design bases and the SSCs which are governed by them are provided in Section C of this Response. The corrective actions relating to SSFE findings and observations, SWOPI plant issues and SSFI deficiencies are discussed in Section D to this Response.

DLC has also conducted two (2) focused Vertical Slice Reviews at BVPS 2 to provide added confidence that the Unit is consistent with its design bases. The selected systems

were the BVPS 2 Auxiliary Feedwater System and Safety Injection System. These focused Vertical Slice Reviews were conducted in accordance with specialized plans that were modeled after NRC Safety System Functional Inspection methodology, but were modified to focus on the design and configuration control aspects of the systems rather than on their operational performance capabilities. Relevant lessons learned from the previously conducted vertical slice reviews were incorporated into the review plan. These reviews concluded that these systems have been maintained consistent with their design bases and that the applicable procedures accurately reflect their design bases.

A.9.2 Vendor Technical Information

DLC developed and implemented a Vendor Technical Information Program in response to NRC Generic Letter 83-28 which is designed to ensure that vendor information related to safety-related SSCs is complete, current and controlled. The Nuclear Engineering Department reviews and approves Vendor Technical Information prior to use. This review is to maintain consistency with existing design bases information. Approval of Vendor Technical Information is documented by the Design Change Package or Technical Evaluation Report processes. These modification processes contain provisions for the translation of appropriate information into operating, maintenance and testing procedures.

A.9.3 Engineering Backlog Reduction

For the past several years the number of Nuclear Engineering Department open work items (work backlog) was identified as a concern because safety significant issues could be overlooked or receive a low priority. In 1996, several projects were initiated whose goals were to reduce the number of backlog items associated with key engineering programs. These engineering programs included Design Changes, Design Equivalent Changes, Engineering Memoranda and Records Update. Collectively, the scope of the projects included :

- completion of records update for operationally accepted or canceled design change packages
- completion of records update for operationally accepted Technical Evaluation Reports
- close-out or rescheduling of Nuclear Engineering Department responses to requested information and/or corrective actions
- verification of select document files for agreement with the document control databases

The backlog reduction effort was effective in reducing the number of in-process Nuclear Engineering Department work items. In addition, engineering responsiveness exhibited an overall improvement. The NRC has also recognized the significant improvements that have been made. The completion of the backlog reduction projects provided added confidence that the site configuration is maintained consistent with the design bases. In

addition, the projects provided lessons learned and identified corrective actions to prevent recurrence. These included the conduct of department training and the revision of administrative procedures.

A.10 PROGRAMMATIC IMPROVEMENTS

The effectiveness of engineering design and configuration control processes is periodically reviewed and assessed through audits, inspections, evaluations and self-assessments. Program changes are made to incorporate self-identified improvements and to correct deficiencies. Over the last five (5) years, the following significant program changes were implemented.

- Technical Evaluation Report Process Improvement Project
- Design Drawing Reconciliations
- Modification Process Improvement Project
- High-Energy Line Break Program
- Electrical Calculation Upgrade Program

In 1995, the Technical Evaluation Report Process Improvement Project was initiated to address self-assessment concerns regarding Technical Evaluation Reports. The project focused on the operational acceptance, configuration control and records updating following modification completion. Department responsibilities associated with Technical Evaluation Report process changes are described in the site administrative procedures.

As a result of discrepancies noted between the UFSAR and other station drawings during NRC Inspection 50-412/94-17, a drawing reconciliation program was implemented. The site administrative procedure governing the program invokes additional controls including the conduct of joint department reviews and approvals of the Valve Operating Number Diagrams and Operating Manual valve lists. Change requests initiated direct the preparer and reviewer to focus on potential impacts to the UFSAR.

The Modification Process Improvement Project was performed in 1994 to address the recommendations of various internal and external reviews. The Modification Process Improvement Project established objectives to improve the overall effectiveness of the Design Change Package process by reducing the modification backlog, the number of tasks and the time needed to complete modifications. The Modification Process Improvement Project implemented other program enhancements such as additional System Engineering involvement in the Design Change Package process, standardization of modification installations and the means for conducting more effective modification testing and improving operational turnover activities.

High-Energy Line Break program improvements were implemented as a result of a reference documentation discrepancy identified in a 1994 NRC Inspection Report

50-334/94-26;50-412/94-27. DLC performed a review to identify the applicable design document containing specific relevant parameters including High-Energy Line Breaks, heat loads and radiation doses. The High-Energy Line Break profile document for each Unit was then revised to include a comprehensive list of design bases calculations supporting these parameters for each applicable building area. These lists serve to enhance the accessibility of design bases information.

The 1991 NRC Electrical Distribution System Functional Inspection resulted in the identification of a number of deficiencies in design documentation, such as Diesel Generator, electrical penetration, relay, and cable sizing calculations. None of the deficiencies resulted in operability issues, however, two (2) of the above deficiencies resulted in the issuance of a Notice of Violation. The majority of the other deficiencies resulted from design documentation in support of calculation assumptions and equipment configurations not being available, particularly for BVPS 1. An appropriately broad corrective action was implemented in the form of an Electrical Calculation Upgrade Program to reconstitute missing or deficient calculations. To date, those calculations which were identified in the original NRC inspection have been upgraded. Other calculation deficiencies are identified, reviewed and corrected on a continuing basis. As a result of performing this design review and reconstitution effort, the electrical distribution calculations were enhanced.

B. RESPONSE TO INFORMATION REQUEST (b)

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

---

B.1 RATIONALE FOR CONCLUSION

DLC's rationale for concluding that design bases requirements have been properly translated into operating, maintenance and testing procedures is based upon established and continually improved procedure change control processes, and the results of routine and special initiative assessment programs, as described in this Section.

When BVPS procedures were developed during the initial licensing of BVPS 1 and BVPS 2, they were based on design basis information and pre-operational and start-up testing activities which demonstrated the Units' readiness and capability to operate safely within their design bases. Since BVPS procedures were developed, comprehensive change control processes have been in effect in accordance with 10 CFR 50, Appendix B. These processes are implemented in accordance with the DLC Operations Quality Assurance Program, and contain attributes designed to assure continuing consistency between procedures and design bases requirements. Controls, such as general and specific procedural provisions, and multi-disciplinary reviews of procedures and design bases requirements by various organizations, are designed to ensure that design bases requirements are translated into operating, maintenance and testing procedures. The procedure development and revision process is a controlled process conducted by a knowledgeable and trained staff.

DLC's implementation of routine, ongoing programs, as well as special initiatives, such as vertical slice reviews, provide additional confidence that design bases requirements are appropriately incorporated into operating, maintenance and testing procedures. In addition, internal and external audits and assessments have provided additional evidence that processes have been effective in maintaining consistency between procedures and design bases. Also, external oversight groups such as the NRC selectively evaluate translation of design requirements into procedures.

As these programs, reviews, and initiatives have identified deficiencies, corrective actions have been or are in the process of being implemented. Collectively, these efforts provide reasonable assurance that design bases requirements are translated into operating, maintenance and testing procedures.

B.2 DESCRIPTION OF PROCESSES

B.2.1 Initial Development

BVPS operating, maintenance and testing procedures have been developed in accordance with the requirements of 10 CFR 50, Appendix B. Since the issuance of the operating licenses for BVPS, procedures have been developed and maintained in accordance with the Operations Quality Assurance Program. BVPS programs relating to the development and revision of operating, maintenance and testing procedures meet NRC Regulatory Guide 1.33 requirements and are continually improved to address deficiencies and emerging new information. Site administrative programs exist which are designed to ensure that design bases and regulatory requirements are appropriately reflected in new and revised procedures.

In the development and use of operating, maintenance and testing procedures, DLC management places emphasis on constantly increasing standards of performance and self-critical responses to emerging issues. A site directive specifies that written procedures, consistent with the UFSARs, Technical Specifications and the Operations Quality Assurance Program, be used to conduct safety-related activities. In addition, site administrative procedures require that site personnel use the latest approved controlled procedures while conducting site activities. Site administrative procedures require that if a procedure cannot be performed exactly or is not available, the work activity is not performed unless required by an emergency or casualty situation.

B.2.2 Procedure Development and Revision

Procedure writer's guides are used for the development or revision of operating, maintenance and testing procedures. Department administrative procedures require procedure writers to review the UFSAR and Technical Specifications in order to make new or revised procedures consistent with applicable design bases requirements. Other reference documents, such as Design Change Packages, Technical Evaluation Reports, Temporary Modification packages, technical manuals, and codes and standards, are reviewed by the procedure writer as appropriate. Procedure writers are trained in the use of these administrative procedures through department-specific training and required reading.

Design Change Packages, Technical Evaluation Reports, Temporary Modifications and Technical Specification amendments are reviewed by procedure writers in the affected departments to determine their impact on operating, maintenance and testing procedures as appropriate. If required after such review, procedures are developed or revised in accordance with the department administrative procedures discussed above. Station personnel can also request that procedures be developed or revised by submitting a change request to the appropriate department.

B.2.3 Procedure Review and Approval

Operating, maintenance and testing procedures developed or revised by procedure writers are independently reviewed. Consideration of the UFSAR and Technical Specifications by both writer and independent reviewer is designed to verify that appropriate design bases requirements are incorporated. Following the independent review, new procedures and procedure intent revisions are screened for 10 CFR 50.59 applicability. If required, a 10 CFR 50.59 safety evaluation is prepared and independently reviewed. The procedures, screening forms and 10 CFR 50.59 safety evaluations are presented to the OSC for review. The OSC reviews the screening forms to determine whether 10 CFR 50.59 safety evaluations have been performed as required. Following an OSC recommendation for approval, the General Manager, Nuclear Operations or his designee approves the procedure.

Station personnel performing and reviewing the 10 CFR 50.59 safety evaluations are required, under DLC administrative procedures, to be trained and certified for such functions. Determining the impact on design bases of a new or revised operating, maintenance or testing procedure is a focus of such training and certification.

10 CFR 50.59 safety evaluations are reviewed by both trained station personnel and the OSC to verify that a new or revised operating, maintenance or testing procedure does not constitute an unreviewed safety question. Prior NRC approval is required for a new or revised procedure that constitutes an unreviewed safety question.

As described in the Technical Specifications, site personnel can perform temporary changes to operating, maintenance and testing procedures provided that:

- The intent of the original procedure is not altered,
- The change is approved by two (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator license on the affected Unit, and
- The change is documented, reviewed by the OSC and approved by the General Manager, Nuclear Operations or his designee within 14 days of implementation.

Field initiated procedure intent changes can be processed by shift personnel, provided that the change to the procedure is reviewed by the OSC and approved by the General Manager, Nuclear Operations or his designee in accordance with the Technical Specifications prior to implementation.

B.2.4 Emergency Operating Procedures

Emergency Operating Procedures (EOPs) are subject to the review and approval process described in Section B.2.3 above, in addition to also being governed by a Procedure Generation Package. This package is applicable to both BVPS 1 and BVPS 2 and was

reviewed by the NRC as documented in the BVPS 2 NRC Safety Evaluation Report, NUREG 1057, Supplement 5.

The original development of EOPs was based on generic Emergency Response Guidelines, supplied by the Westinghouse Owners Group[4]. The EOPs were generated by applying BVPS specific parameters to the guidelines and reconciling the BVPS specific procedures with the UFSAR and plant design bases. Pursuant to the Procedure Generation Package, any new or revised EOP which is not initiated by the Westinghouse Owners Group must be reviewed against source documents, including applicable design bases requirements, Westinghouse Emergency Response Guidelines and the UFSARs. This review is conducted to verify that the approved accident mitigation strategy is maintained. Revisions initiated by the Westinghouse Owners Group are controlled by the Westinghouse Quality Assurance Program, but also receive applicable UFSAR design bases reviews performed by DLC personnel.

A validation is required to be performed for new or revised EOPs involving changes in intent, setpoints, step sequence, or referencing and branching. The validation consists of either a table-top, walk-through, simulator demonstration, or a combination of these methods. An independent review is performed for EOP revisions. The subsequent review and approval process includes 10 CFR 50.59 screening, OSC review and approval by the General Manager, Nuclear Operations or his designee.

B.2.5 Procedure Issuance and Control

After an operations procedure or procedure revision is approved, it is forwarded to the Beaver Valley Records Center for controlled distribution to appropriate locations. Approved new or revised maintenance and testing procedures are maintained in controlled master files. To ensure use of the latest revision, working copies of maintenance and testing procedures are required to be obtained from the original master files.

B.3 TRANSLATION OF DESIGN INFORMATION INTO PROCEDURES

Site administrative procedures require that new or revised design information be transmitted to appropriate site organizations for incorporation into operating, maintenance and testing procedures. As discussed in Section B.2 above, the procedure writer is required to review this design information and apply it to existing operating, maintenance and testing procedures to determine the need for new procedures and/or procedure revisions. Review checklists used by the procedure writer during the procedure development and revision process require the review of numerous items, including Technical Specification requirements, acceptance criteria, operating data, required values, technical guidance, system and component interactions, commitments and references.

---

[4] An organization of utilities, both foreign and domestic, which own nuclear power plants with Westinghouse-designed Nuclear Steam Supply Systems which addresses generic engineering, licensing and operational issues.

New or revised procedures are independently reviewed to verify that design bases requirements have been incorporated.

Operating, maintenance and testing surveillance procedures include acceptance criteria that satisfy the surveillance requirements of Technical Specifications, as well as the ASME In-Service Testing program, the Offsite Dose Calculation Manual and balance of plant testing. Design bases requirements are reflected in the surveillance requirements established in the Technical Specifications, and these design bases requirements are translated into the acceptance criteria of surveillance procedures through the processes described in Section B.2. Satisfactory completion of surveillance tests provides confidence that an SSC can perform its intended safety function.

Vendor Technical Information is controlled in accordance with site procedures. Existing Vendor Technical Information is obtained from the Beaver Valley Records Center for use in the development or revision of operating, maintenance and testing procedures. The Nuclear Engineering Department is responsible for evaluating and authorizing Vendor Technical Information prior to it being forwarded to the Beaver Valley Records Center.

When Vendor Technical Information is received from sources other than the Beaver Valley Records Center, a request for engineering evaluation and approval of the information is generated in accordance with site procedures. The use of Vendor Technical Information is not authorized without prior engineering approval. The Nuclear Engineering Department approval of Vendor Technical Information is documented by the Design Change Package or Technical Evaluation Report processes. These modification processes contain provisions for translating appropriate Vendor Technical Information into site operating, maintenance and testing procedures.

B.4 ASSESSMENT PROGRAMS

B.4.1 Routine Assessments

The procedure writer, the independent reviewer and the procedure writer's supervisor are responsible for ensuring that design bases information is properly translated into operating, maintenance and testing procedures. The Quality Services Unit provides additional assurance by conducting routine assessments to monitor and verify that design bases information is properly translated. The Quality Services Unit implements a comprehensive system of planned, periodic audits that includes verifying that the operating, maintenance and testing procedure control processes are properly implemented. Although deficiencies have been identified, Quality Services Unit audit results show that the translation of design bases requirements into operating, maintenance and testing procedures has been acceptably performed.

In addition to the routine Quality Services Unit assessments, routine departmental self-assessments are also conducted to evaluate the implementation of procedure control processes. An example of this is an assessment performed by the Operation Procedures Section that concerned a Problem Report which identified emergency powered 480VAC

power supplies for the alternate intake structure intake fans that were not addressed in the appropriate systems' power supply list. The master 480VAC emergency load list was cross-referenced with each system power supply list to verify that emergency loads were included. The review concluded that five (5) out of approximately nine-hundred (900) 480VAC emergency loads were not included in the systems' power supply list. The missing loads, however, were contained in the master load list, and would be verified in the correct position during post-outage lineups. The system power supply lists were corrected. Based on the number of discrepancies found, no further action was recommended.

B.4.2 Special Initiatives

Vertical slice review results provide another basis that corroborates the effectiveness of the processes for accurately translating the design bases into operating, maintenance and testing procedures. As discussed in Section A, DLC conducted vertical slice reviews on ten (10) safety-related systems at BVPS 1 and one (1) safety-related system at BVPS 2 to verify that they are capable of performing their intended safety functions under normal operating and accident conditions.

The BVPS 1 systems were:

- Auxiliary Feedwater System (1988)
- Quench Spray System (1988)
- Emergency Diesel Generator (1988)
- Reactor Plant River Water System (1989)
- Recirculation Spray System (1989)
- Residual Heat Removal System (1989)
- Supplementary Leak Collection and Release System (1990)
- Electrical Distribution System (1991)
- River Water System (SWOPI) (1994)
- Safety Injection System (SSFI) (1995)

The BVPS 2 system was the Service Water System (SWOPI) (1994).

Each vertical slice review specifically included an assessment of whether design bases requirements are translated into operating, maintenance and testing procedures. A checklist was developed and used to specifically verify that design bases requirements were incorporated into selected procedures.

Of the issues identified by the SSFEs that related to the translation of design bases requirements into procedures, five (5) were initially determined to be significant and were classified as findings. These were: (1) inconsistent post-maintenance dynamic stroke

testing of Motor Operated Valves; (2) lack of calibration and functional testing of flow indicating switches which indicate Quench Spray Pump discharge; (3) questions concerning the acceptability of the configuration used to test the Design Basis Accident flow conditions for the Recirculation Spray System; (4) lack of a bypass leakage test on the Supplementary Leak Collection and Release System Main Filter Banks; and (5) lack of surveillance testing of the cross connect valves between the oil fuel storage tanks for the Emergency Diesel Generators.

Motor Operated Valve testing was determined to be adequate based on additional testing information. For the remaining four (4) findings, additional tests and analyses showed that the systems and components met design bases requirements. For the Recirculation Spray System, the test configuration was determined to be adequate and the UFSAR was modified accordingly.

Of the plant issues identified by the SWOPI, two (2) involved the adequacy of plant procedures and testing to incorporate design bases information. One involved heat exchanger testing and monitoring, and the other concerned intake bay cleaning, and clam and corrosion control.

SWOPI corrective actions included enhancements to heat exchanger testing and monitoring to improve testing methodology, including test acceptance criteria, monitoring frequency and as-found assessments. Overall responsibility for the Asiatic clam control program was assigned to River Water and Service Water System Engineers, to better coordinate related maintenance.

Of the deficiencies identified by the SSFI, four (4) were related to the translation of design requirements into procedures and were significant enough to generate problem reports. These were: (1) lack of in-service testing of two check valves used to isolate the Boron Injection Recirculation Pump discharge; (2) non-conservative discharge pressure requirement for Charging Pump Technical Specification surveillance; (3) lack of surveillance testing of the automatic Safety Injection block function; and (4) lack of testing of the latch-in feature of the Reactor Protection System auto recirculation output relays.

SSFI corrective actions included submittal of a Technical Specification change request to correct the non-conservative discharge pressure value. In each case, the applicable procedures were revised to address the identified deficiency.

These vertical slice reviews also identified several other less significant issues. Their number and nature are consistent with the vintage of BVPS 1. These issues were evaluated for reportability and operability, assigned to responsible departments for closure, tracked and corrected. Actions to prevent recurrence included revisions to and enhancement of various plant programs and procedures.

In conclusion, the design bases translation issues discovered by these vertical slice reviews showed that the design translation process was effective and provided additional

assurance that design bases requirements are translated into operating, maintenance and testing procedures. Additional details about the vertical slice reviews conducted at BVPS are provided in Sections A, C, D and E of this Response.

As discussed in Section A of this Response, in 1996, DLC also conducted two (2) focused Vertical Slice Reviews of the BVPS 2 Auxiliary Feedwater and Safety Injection systems. The review identified one (1) potential observation related to the application of the 10 CFR 50.59 process. A DLC evaluation of this observation determined that the 10 CFR 50.59 process was correctly performed, and therefore, no further corrective action was necessary. These reviews also assessed translation of design bases requirements into selected procedures. These reviews concluded that the design bases of the selected systems have been adequately maintained and were reflected in the operating, maintenance, and testing procedures.

In response to a design deficiency identified in 1996 with the BVPS 1 Anticipated Transient Without Scram Mitigating Actuation Circuitry System (AMSAC), DLC initiated a set of four (4) Focused Design Reviews. These reviews were performed on BVPS 1 subsystems that were added to the plant as Design Change Packages. The purpose of these reviews was to evaluate the functional adequacy of the as-built design for each system. The four (4) Focused Design Reviews were specifically directed to verify or ascertain that operating, maintenance and testing procedures accurately reflect design bases requirements. The reviews identified two generic issues related to translation of design bases into procedures. The first involved either deficiencies in initial testing or the lack of adequate documentation of existing testing. The second generic issue involved the absence of a process to confirm that periodic field calibration results regarding instrumentation drift meet design bases assumptions. Both issues are currently under review to develop corrective actions to prevent recurrence.

While deficiencies have been identified as described above, the audits, inspections, assessments and reviews have confirmed the integrity of the procedure control process and the translation of design bases requirements into operating, maintenance and testing procedures.

B.4.3 External Oversight

External oversight groups such as the NRC conduct assessments, inspections and evaluations on a routine basis. These activities selectively assess implementation of the procedural control process, and translation of design bases requirements into procedures.

B.5 CONCLUSION

The controls, programs, special initiatives, and internal and external oversight discussed above give DLC reasonable assurance that design bases requirements are translated into operating, maintenance and testing procedures. Where deficiencies have been identified, corrective actions have been or are in the process of being implemented.

C. RESPONSE TO INFORMATION REQUEST (c)

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

---

C.1 RATIONALE FOR CONCLUSION

DLC's rationale for concluding that the configuration and performance of the SSCs in each Unit of BVPS are consistent with their design bases is supported by a number of factors. First, each Unit was subjected to a start-up testing program that was developed and carried out before each Unit began commercial operation. Each initial testing program provided the original basis for reasonable assurance, as that criterion was applied when each of the Units was licensed, that the configuration and performance of the SSCs for each Unit are consistent with the design bases.

Second, since each Unit began operation, the configuration and performance of its SSCs have been maintained consistent with their design bases through routine operation, maintenance and modification activities, and the implementation of controlled, comprehensive procedures and processes. These procedures and processes (which are described in this Section and in Sections A, B and D of this Response) control design changes and operating, maintenance and testing activities, and, in particular, include steps for the use of design bases information.

Third, DLC relies upon ongoing performance testing (and other routine programs) to verify that SSCs are maintained, and perform, within their design bases.

Fourth, audits, assessments and other oversight activities have also been conducted.

Finally, in recognition that BVPS 1 was licensed before the TMI incident, DLC has undertaken special initiatives at BVPS 1 to provide ongoing confirmation that SSC configuration and performance are consistent with its design bases. These special initiatives include a series of vertical slice reviews, various configuration verification initiatives, and a series of Focused Design Reviews. It is also recognized that for BVPS 2, as with other post-TMI plants, the scope of the start-up testing program, regulatory requirements, expectations for the availability of design bases information and the comprehensiveness of design change processes were more rigorous than those that were applied to BVPS 1 and other pre-TMI plants. Two (2) focused Vertical Slice Reviews at BVPS 2 have also been performed. These programs and reviews have identified deficiencies. None was significant enough to challenge a finding of reasonable assurance that BVPS SSCs are configured and perform consistent with their design bases.

Collectively, these activities, processes, testing programs, inspections and initiatives as well as the analysis of recent plant events provide reasonable assurance that SSC configuration and performance at BVPS are consistent with the facility's design bases.

This Section first describes the initial start-up testing programs at BVPS 1 and BVPS 2. Next, the Configuration Management Program, which is comprised of various processes, is discussed, followed by a summary of performance testing programs. Other configuration-related initiatives, including the series of vertical slice reviews, other regulatory programs and walkdowns, Focused Design Reviews, and self-assessments are then described. Finally, the analysis of nuclear industry experience as an additional mechanism for maintaining configuration control is discussed.

C.2 INITIAL START UP TEST PROGRAMS

Before BVPS 1 began operation in 1976, an initial test program was established and implemented, and site personnel developed operating, maintenance, and testing procedures. A joint test group, comprised of senior site personnel and representatives from Westinghouse Electric Corporation and Stone and Webster Engineering Corporation, reviewed and approved procedures to ensure that FSAR commitments, applicable codes and standards, and the applicable 10 CFR requirements were addressed. The BVPS 1 Program followed 1970 AEC Guides, "Guide for the Planning of Pre-operational Testing Programs," and "Guide for the Planning of Initial Startup Programs." Start-up testing provided reasonable assurance that safety-related SSCs (and various non-safety related SSCs) performed consistent with their design criteria and safety function.

The BVPS 2 Program was developed under the guidance of Regulatory Guide 1.68 (Rev. 2), "Initial Test Programs for Water Cooled Nuclear Power Plants." The administrative requirements for the test program were defined in the BVPS 2 Start-up Manual. These procedures required the use of controlled design information and identified the process for preparing, reviewing and approving test procedures for use at BVPS 2. In 1987, administrative procedures were reviewed and a joint BVPS 1 and BVPS 2 administrative manual was created. The successful completion of these activities provided reasonable assurance that the SSCs met their design bases.

In March 1983, during the licensing phase of BVPS 2, DLC established a Design Bases Endorsement Program which was conducted under the DLC Construction Quality Assurance Program using approved procedures and qualified engineering personnel. The scope included a review of design bases documents, which led to DLC's endorsement of the Stone & Webster Engineering Corporation design criteria documents and confirmation of their design process. The program also included a review of specifications, drawings, and licensing documents including the FSAR and a validation of key attributes that addressed the design criteria, design process and design controls.

The NRC reviewed and/or audited the Design Bases Endorsement Program and the Engineering Confirmation Program on four (4) occasions between May 1984 through June 1985. The NRC found the manpower commitment to this program to be substantial

with an expenditure of over 11,000 man hours by forty-eight engineers. DLC had no unresolved concerns with the BVPS 2 design bases but did identify some specific design discrepancies. These discrepancies were subsequently resolved with the Stone & Webster Engineering Corporation through the BVPS 2 Design Bases Endorsement Follow-On Program.

In November 1985, the NRC issued a request for information asking DLC to identify actions taken to assure that BVPS 2 was designed in accordance with FSAR commitments and NRC regulations. DLC responded by taking credit for the existing BVPS 2 Design Bases Endorsement Program combined with the project Engineering Assurance Program in lieu of a pre-operational Integrated Design Inspection or Independent Design Verification Program. Additional reviews were conducted that included NRC oversight. This response was accepted by the NRC and the issue was closed on April 4, 1987.

C.3 CONFIGURATION MANAGEMENT

For BVPS 1, the Configuration Management Program evolved as industry and NRC expectations changed, especially after the TMI incident. After BVPS 2 was licensed to operate, an integrated program applicable to both Units was adopted. It takes into account design requirements and the implementation of those requirements, and controls changes to SSCs. The Configuration Management Program also includes elements that are intended to assure that design documents accurately reflect the as-built facility and operating conditions at BVPS. In this way, the Configuration Management Program helps to ensure that SSC configuration and performance remain consistent with the design bases. The basic elements of the Configuration Management Program have already been described in Section A.

Details about the processes important to configuration management are discussed below. They include system configuration controls, performance testing and surveillance, configuration and performance verification initiatives, self-assessments, and industry experience.

C.3.1 System Configuration Controls

System configuration controls are established to maintain the system configurations consistent with their design bases. These processes are proceduralized and administratively controlled. They include procedures and processes for Normal System Arrangement, Shift Tours and Logs, Equipment Clearances, work control processes, component identification, equipment lists, material management, Maintenance Work Requests and Post-Maintenance Testing. These processes are described in the following sections.

C.3.1.1 Normal System Arrangement

BVPS operating procedures require that system configurations be consistent with Normal System Arrangement. The Normal System Arrangement of a system is the operational alignment of valves, breakers, instrumentation and components that is considered to be normal for unit operation. System configurations are established by performing alignments using Valve, Power Supply and Control Switch Lists. Normal System Arrangement deviations, except for those controlled by clearances or procedures, are recorded in the Normal System Arrangement Deviation Review Log. The BVPS Operating Manual provides instructions for incorporating changes to the Normal System Arrangement and verifying that a system is configured properly.

The Nuclear Operations Unit performed two (2) self-assessments related to Normal System Arrangement. The first self-assessment, performed in 1995, identified discrepancies between the Valve Operating Number Diagram control room status prints and the Normal System Arrangement. Corrective actions taken to resolve these discrepancies included the return of valves to Normal System Arrangement. The second self-assessment, performed in 1996, identified inconsistencies in the use of the Normal System Arrangement Deviation Review Log. Actions taken to correct these discrepancies included updating the Normal System Arrangement Deviation Review Log and the associated Valve Operating Number Diagrams. Corrective actions to prevent recurrence are in the process of being implemented. These actions include retraining on the Normal System Arrangement procedure and periodic audits of the process. Additionally, a surveillance procedure was developed to review the Normal System Arrangement Review Log to provide additional assurance that equipment is properly controlled.

The problem reporting system also identified a problem in ensuring that Operations is notified of Normal System Arrangement changes. A new process designed to ensure that changes are identified and physically completed via a signed notification form was developed. In 1996, a self-assessment of the process was performed over a 9-month period by the Operation Procedures Section. The assessment determined that the process was effective in ensuring Normal System Arrangement changes were properly configured.

A project to improve the distribution of Operator Aids was performed to ensure that the correct revision is located in the field. This project was initiated in response to an NRC violation on Operator Aids. Following implementation of the new process, an evaluation was performed and it was determined that it was effective in providing the field with current design bases information.

C.3.1.2 Shift Tours and Logs

Operations personnel perform routine tours on a shift or daily basis to verify proper plant operation and performance, and to identify degrading trends and material condition. Essential equipment is checked for safe and efficient operation. Pertinent data is

collected by the operations staff and recorded on log sheets. Deviations and corrective actions are also noted in these logs.

C.3.1.3 Equipment Clearances

Equipment clearance procedures are designed to provide administrative control for the removal from service and return to service of equipment. The clearance procedure requires that signed, approved forms be used to detail the specific equipment manipulation. This provides additional assurance that the Normal System Arrangement is maintained following maintenance activities. For Engineered Safety Feature equipment clearances, an Engineered Safety Feature Checklist is completed to ensure that redundant train components are available and administrative requirements are completed.

C.3.2 Work Control Processes

Processes have been established with provisions to restore SSCs to pre-work conditions and to prevent unauthorized Design Changes from occurring during the performance of work. These include the proper identification of equipment and materials, review of work scope prior to the start of work, use of approved procedures and processes, approval of the work activity by the Nuclear Shift Supervisor, and functional testing of equipment following the completion of work. These processes provide additional assurance that the configuration and performance of the SSC remain consistent with the design bases.

C.3.2.1 Component Identification

Component identification processes provide additional assurance that site personnel properly position, operate, and maintain the correct SSC configuration. Site administrative procedures provide guidance for controlling the site tagging and labeling program. This program has provisions for standard labeling with proper nomenclature and Equipment Identification Number, temporary labels, color coding for emergency actuation components, and guidelines for identifying and replacing missing tags.

C.3.2.2 Equipment Lists

Several equipment lists help to maintain the configuration of SSCs consistent with their design bases. Chief among these is the Master Equipment List, a computer database that is used to identify and maintain component configuration control. It provides the link between the engineering design attributes, change documents, and other information necessary to maintain the installed configuration. Site personnel have ready access to the list's design bases information that applies to their work activities. Data in the Master Equipment List is administered by the Nuclear Engineering Department and maintained by the data field owners.

In addition to the Master Equipment List, other controlled programs are used to document SSC configuration. These programs include the Environmental Qualification Master List, the 10 CFR 50, Appendix R (for BVPS 1) and Safe Shutdown Reports (for

BVPS 2), Setpoint Control Program and the Regulatory Guide 1.97 (Post Accident Monitoring) instruments list.

C.3.2.3 Material Management

DLC has established a material management process to control material used at the site. Safety-related parts are procured to the requirements of 10 CFR 50, Appendix B. The Nuclear Procurement Department administratively controls a material reservation system that is available to site personnel. The Nuclear Procurement Department evaluates new material requests by reviewing design documents to determine suitability. The Nuclear Procurement Department also evaluates the use of commercial grade material in safety-related applications.

C.3.2.4 Maintenance Work Requests

Maintenance Work Requests are used to control maintenance work activities. Maintenance Planning and Administration Department reviews Maintenance Work Requests before work is performed to check that no unauthorized plant modifications or design changes result from the work activity. The Maintenance Planning and Administration Department is required to use controlled drawings, procedures, and technical manuals to ensure that the plant configuration is maintained. Plant Installation Process Standards are available to provide generic standards and guidelines for specific installations. In addition, Quality Services Unit personnel review Maintenance Work Requests to identify inspection requirements. Quality Services Unit personnel also review maintenance procedures for appropriate quality control hold points. The on-shift Nuclear Shift Supervisor is required to approve a Maintenance Work Request prior to the initiation of work activities.

C.3.2.5 Post-Maintenance Testing

Another check on configuration control and performance is provided by post-maintenance testing. Site procedures require that appropriate post-maintenance testing be performed following the completion of maintenance work and prior to equipment being considered operable. The Nuclear Shift Supervisor specifies post-maintenance testing requirements. The Nuclear Shift Supervisor also reviews the appropriate Post-Maintenance Testing recommendations from the Maintenance Programs Unit and the System and Performance Engineering Department. Post-maintenance testing provides a means to ensure that the SSC configuration and performance remain consistent with the design bases.

C.4 PERFORMANCE TEST PROGRAMS

The performance of SSCs is verified and maintained by various programs and processes, which include testing, inspection and oversight activities. Testing programs (surveillance, modification, In-Service Testing, etc.) and inspection programs (Equipment Qualification, In-Service Inspection, etc.) are designed to evaluate and verify SSC

performance and to identify deficiencies. The conduct of these activities also helps to maintain the performance of SSCs consistent with their design bases. Other related activities, such as System Engineering, Maintenance Rule and predictive maintenance programs, also provide additional assurance that BVPS SSCs are being operated and maintained in accordance with design bases requirements. Collectively, the implementation of these programs and processes provides additional assurance that the performance of the SSCs is consistent with the design bases.

C.4.1 Surveillance Testing

Design bases and performance requirements are reflected in the surveillance tests established by the BVPS Technical Specifications and regulatory requirements. The satisfactory completion of surveillance tests demonstrates that the SSC can perform its intended safety function. Design bases requirements are incorporated into the surveillance test acceptance criteria. Tests that implement these surveillance requirements include Operating Surveillance Tests, Beaver Valley Tests, Reactor Surveillance Tests, Maintenance Surveillance Procedures, and shift operating logs. The Nuclear Shift Supervisor performs operability and reportability determinations for deviations from test acceptance criteria. SSCs that do not meet test acceptance criteria are considered to be inoperable and corrective actions are initiated.

C.4.2 Modification Testing

Modification testing is intended to provide assurance that the SSC as modified will perform consistent with the design bases requirements. Site administrative procedures establish requirements, responsibilities, and guidelines for the Design Change Package modification testing processes. This process provides a test plan detailing functional and operational acceptance testing requirements based on SSC design bases information. The Maintenance Support Engineering Section coordinates the implementation of the Test Plan and reviews the results. This review is intended to ensure that the testing has been satisfactorily completed in accordance with the Test Plan requirements and acceptance criteria.

C.4.3 In-Service Inspection Program

The Nuclear Engineering Department administers the DLC In-Service Inspection Program, the function of which is to maintain the integrity of safety-related fluid systems and components in accordance with 10 CFR 50.55a. The program also incorporates the requirements of the BVPS 1 and BVPS 2 UFSARs. The program is based on physical examinations of plant components. Deficiencies identified during these inspections are resolved by repair, replacement, or engineering evaluation prior to returning the component to service. In-Service Inspections provide an added measure of assurance that the SSCs inspected are consistent with the design bases.

C.4.4 In-Service Testing

Additional evidence that SSCs are performing consistent with their design bases is provided through the In-Service Test Program, which is administered by the System and Performance Engineering Department. The In-Service Testing Programs at BVPS 1 and BVPS 2 follow the guidelines of ASME Section XI and the positions in Attachment 1 of Generic Letter No. 89-04, "Guidance on Developing Acceptable In-service Testing Programs," including Supplement 1 (NUREG-1482, "Guidelines for In-Service Testing at Nuclear Power Plants"), to verify operational performance of the In-Service Testing of pumps and valves in accordance with 10 CFR 50.55a. The tests used for the In-Service Testing Programs include Operations Surveillance Tests, Beaver Valley Tests, Corrective Maintenance Procedures, and Surveillance Logs. A self-assessment was conducted during 1994 to verify that the program's administrative and implementing procedures met regulatory requirements. This assessment identified a check valve flow testing concern, which was subsequently corrected. The program is currently being revised for the next 120-month ASME Section XI Code interval.

C.4.5 Motor-Operated Valve Testing

In response to Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," DLC developed a program designed to test, inspect and maintain MOVs to provide the necessary assurance that they will function as required when subjected to design bases conditions. In order to implement the MOV Program, relevant MOV and system operating characteristics were reviewed, verified and reconstituted when necessary. An NRC review performed in 1995 concluded that the program addressed the requirements of the Generic Letter. Although this review identified strong management support of the program, it also identified fundamental errors in the MOV test data analysis methodology and switch setting calculations, along with inadequate justifications for technical details. Prior to the end of this review, actions were taken to correct the calculation errors and to provide sound assumption justifications, thus providing closure to the Generic Letter.

C.4.6 Predictive Maintenance Programs

In addition to the testing programs described above, other equipment performance programs have been established at BVPS to improve the overall reliability of plant equipment, including SSCs. These programs are not regulatory in nature, but are designed to ensure that equipment is reliable and is performing consistently within established criteria. These programs include:

- Air Operated Valve Program - A program to maintain the performance margin and reliability of air-operated valves.
- Electrical Protection Program (EPRI NP 7216) - A program to maintain and monitor the performance of electrical protection and control relays.

- Vibration Monitoring Program (EPRI NP 7205) - A program to perform periodic vibration checks on major plant rotating equipment and trend the results to use as an early warning indicator of equipment problems.
- Lubrication Sample Analysis Program (INPO MA 316) - A predictive maintenance program involving the collection and analysis of lubrication. The results of the analysis are trended and used as an indicator of equipment condition.
- Thermography Program (EPRI NP 6973) - A predictive maintenance program involving measuring the thermal temperatures of various components to determine performance, and as an indicator of equipment condition.

C.4.7 Performance Oversight Programs

Various performance oversight programs provide additional assurance that the SSCs' configuration and performance are consistent with their design bases. These programs provide an overview of SSC operating characteristics, material condition and availability.

C.4.7.1 System Engineer Program

In 1992, DLC established a System Engineering Program. The System Engineers are assigned to systems in order to provide an integrated approach to system operation, maintenance and engineering. System Engineers are required to be qualified for a system by being familiar with its design and configuration. Responsibilities of the System Engineer include implementation of the Maintenance Rule (including availability trending and Maintenance Preventative Functional Failure review), Preventative Maintenance Program optimization, twelve-week schedule input, performance trends, and field walkdowns for each assigned system. The System Engineer provides oversight of system availability and effectiveness with respect to design bases performance. Through the performance of periodic walkdowns, the System Engineer retains an informed overview of system operating characteristics, material condition, equipment problems/discrepancies, and general compliance with plant procedures. Potential conditions adverse to quality are evaluated and brought to the attention of appropriate station management, normally via Condition Reports (described in Section D of this Response).

C.4.7.2 Maintenance Rule Program

The purpose of the Maintenance Rule Program is to monitor the effectiveness of maintenance activities on key SSCs (both safety and non-safety related) in accordance with the requirements of 10 CFR 50.65 and Regulatory Guide 1.160. The implementation of the Maintenance Rule Program involves monitoring the performance of SSCs to provide additional assurance that they are capable of fulfilling their design bases requirements. Administrative procedures were developed to control the implementation of the program. When the Maintenance Rule Program was initially established, DLC identified the function of SSCs within the scope of the rule utilizing

design bases information. Performance criteria and risk significance were developed based on these functions. System Engineers monitor these criteria to ensure that the SSCs within the scope of the rule meet their performance criteria. A self-assessment of this program identified the need for program improvements. These improvements were developed and have been incorporated into the current program.

C.5 CONFIGURATION AND PERFORMANCE VERIFICATION INITIATIVES

The following initiatives provide additional assurance that the programs and processes described in this Section, as well as those described in Sections A and B, have been properly implemented with respect to SSC configuration and performance. These initiatives include vertical slice reviews, Focused Design Reviews, responses to Generic Letter actions and I. E. Bulletins, Operations Department assessments, and industry experience.

C.5.1 Vertical Slice Reviews

Vertical slice review results provide additional confidence that the configuration and operation of SSCs are consistent with their design bases. As discussed in Sections A, B, D and E of this Response, DLC conducted vertical slice reviews on ten (10) safety-related systems at BVPS 1 and one (1) safety-related system at BVPS 2 to verify that they are capable of performing their intended safety functions under normal operating and accident conditions.

The BVPS 1 systems were:

- Auxiliary Feedwater System (1988)
- Quench Spray System (1988)
- Emergency Diesel Generator (1988)
- Reactor Plant River Water System (1989)
- Recirculation Spray System (1989)
- Residual Heat Removal System (1989)
- Supplementary Leak Collection and Release System (1990)
- Electrical Distribution System (1991)
- River Water System (SWOPI) (1994)
- Safety Injection System (SSFI) (1995)

The BVPS 2 system was the Service Water System (SWOPI) (1994).

The SSFE vertical slice reviews made many comparisons between SSCs and their design bases. The bulk of those comparisons showed agreement between the SSCs and their design bases. The SSFEs also identified deficiencies in both the configuration and performance areas.

Of the configuration issues identified by the eight (8) SSFEs, eleven (11) were significant enough to be classified as findings. These were: (a) a breach of fire seals between the AFW pump area and Fire Zone CV-1; (b) deactivation of three (3) MOV valve operators; (c) inadequate wiring configuration/isolation for two level transmitters; (d) a non-seismically supported roof drain line inside the EDG building; (e) improper use of a PVC pipe in the Auxiliary Building; (f) reach rod design and configuration deficiencies; (g) inadequate pressure ratings for a valve and two (2) flanges that mated MOVs; (h) inadequate administrative protection for access and electrical separation for offsite power lines in the switchyard; (i) deficient fuel oil system missile/tornado protection; (j) incorrect positioning of circuit breakers for the 4kV swing pumps; and (k) incorrect EDG fuel oil tank low level setpoints.

One issue, breach of fire seals, was determined to be an acceptable outage configuration for use of test equipment cables. For four (4) of the other findings, additional analysis confirmed the acceptability of the as-found conditions. Four (4) other findings required some plant activity (e.g., removal of the PVC pipe, removal of one (1) set of 4 kV swing pump breakers) and updating of design documentation. For the remaining two (2) findings (MOV valve operators and level transmitter wiring configuration), a Design Change Package was implemented to address the identified deficiency.

One (1) performance-related SSFE finding (not previously discussed in Section B) was related to torque switch settings for an MOV. In this finding, MOV torque settings were found to be overly conservative. The settings were subsequently changed to improve valve reliability.

For the SWOPI, of the five (5) plant issues identified, none were directly associated with plant configuration.

Overall system performance of the BVPS 1 River Water and BVPS 2 Service Water Systems was found to be acceptable by the SWOPI. Three (3) performance issues were characterized by the SWOPI as plant issues: piping degradation, system margins and inadequate freeze protection. These issues have been adequately addressed via program enhancements which were coordinated through the efforts of the River/Service Water Task Force. Remedial actions included changes to the River/Service Water Inspection Monitoring Program and Design Change Controls to improve (1) system flow margins which included elimination of a BVPS 1 River Water cross connection and (2) adequate freeze protection control at the alternate intake structure. An NRC inspection was performed in 1995 which concluded that these corrective actions were consistent with the Generic Letter 89-13 requirements. A Quality Services Unit assessment performed in 1996 concluded that these corrective actions were being completed as scheduled.

Of the configuration deficiencies identified by the SSFI, one was determined to be significant enough to generate a Problem Report. This deficiency identified that a pipe hanger for a Safety Injection line was not installed. An analysis was performed of the piping system without the support in place and concluded that the piping system met the piping design criteria.

Of the performance deficiencies identified by the SSFI, two (2) were determined to be significant enough to generate Problem Reports. These were : (1) flow element errors in instruments used for balancing the High Head Safety Injection cold and hot leg branches; and (2) the effects of Emergency Diesel Generator frequency variations on the High Head Safety Injection Pump performance. For the flow element error deficiency, an analysis was performed that determined the errors did not exceed the Safety Injection System design bases. For the frequency variation deficiency, an analysis was performed that determined the expected variation in the Emergency Diesel Generator frequency would not produce unacceptable flow conditions for the Safety Injection System. No plant modifications were required to close these issues.

These vertical slice reviews also identified several other less significant configuration and operation issues. The number and nature are consistent with the vintage of BVPS 1. These issues were evaluated for operability and reportability, assigned to the responsible departments for closure, tracked and corrected.

Two (2) focused Vertical Slice Reviews conducted at BVPS 2 provide additional confidence that its SSCs are maintained and perform in accordance with design bases requirements. In 1996, an experienced independent review team used a vertical slice sampling approach to review records of selected activities that affected or were affected by the design bases for the Auxiliary Feedwater System and the Safety Injection System at BVPS 2. The objective of these focused Vertical Slice Reviews was to form a conclusion as to whether the design bases for these systems were being adequately maintained. System walkdowns were performed on selected portions of the two systems, and discussions were held with plant support staff to facilitate the review.

The team identified a number of potential issues related to the design control process and design documentation. DLC evaluated these issues for operability considerations. No operability issues were identified. The most significant issues were; (1) failure to update a BVPS 2 UFSAR drawing following a Design Change, (2) lack of a 10 CFR 50.59 Safety Evaluation for a change in the Normal System Arrangement procedural control of a Chemistry Sample Valve, and (3) the installation of an equivalent replacement part without preparing adequate modification documents and updating associated design documents. The DLC assessment determined that the part was acceptable for use. Condition Reports were issued, and corrective actions for these three (3) issues are in the process of being implemented.

The team concluded that the design bases of the BVPS 2 Auxiliary Feedwater System and Safety Injection System have been adequately maintained. Principal design information for both the Auxiliary Feedwater System and Safety Injection System was found to be accurate and consistent. The team concluded that the design information was reasonably well-organized and accessible for users familiar with the document control and retrieval systems.

The focused Vertical Slice Review team also found that over the interval from BVPS 2 licensing to the present, based on the documents reviewed, there has been a significant

improvement in the formality, rigor and documentation of activities that maintain the design bases and control the configuration of the facility.

C.5.2 Focused Design Reviews

In response to a design deficiency identified in 1996 with the BVPS 1 AMSAC system, DLC initiated a set of four (4) Focused Design Reviews. These reviews were performed on BVPS 1 subsystems that were added to the plant as Design Change Packages. The purpose of these reviews was to verify or ascertain that the existing procedures and as-built designs accurately reflected the design criteria and compliance with the applicable NRC regulations. The selected systems were AMSAC, Inadequate Core Cooling Monitoring / Reactor Vessel Level Indication, High-Energy Line Break Isolation, and the Post Accident Sampling System. The Focused Design Reviews reached the conclusion that the specific systems reviewed were functional. Recommendations to enhance reliability were also provided.

Corrective actions for the issues identified have been assigned and are currently in progress. The Focused Design Review results were also reviewed for generic implications. Two generic issues were identified. These were related to weaknesses in functional testing and the review of setpoint drift compared to calculations. Two (2) corrective actions are under development to prevent recurrence. The first is a Quality Services Unit follow-up review on functional testing. The second is to develop a method to review as-found calibration data. Both actions have been assigned and entered into the site commitment tracking system for disposition.

C.5.3 Equipment Qualification Program

In accordance with the requirements of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety at Nuclear Power Plants," the Nuclear Engineering Department has established an Equipment Qualification Program, which is designed to ensure that electrical equipment important to safety remains operable during a Design Basis Accident. The Nuclear Engineering Department maintains a list (Electrical Equipment Qualification Master List) of such equipment for each Unit. The relevant design documentation (e.g. operating and environment parameters, qualification testing requirements and results, aging assessments, life-cycle data) for each equipment item on these lists is maintained in separate controlled Equipment Qualification files available in the BVRC. The Nuclear Engineering Department also specifies and tracks the maintenance requirements for the equipment. These requirements, including periodic inspections, replacements and surveillance testing, provide additional assurance that the designated equipment continues to meet design bases requirements. In 1988, two Environmental Qualification deficiencies resulted in a NRC Severity Level IV Notice Of Violation. The first issue was related to the use of unqualified electrical cable in a radiation monitoring system. The second issue involved the use of unqualified jumpers in motor operated valves. Both of these deficiencies were corrected. Follow-up actions to prevent recurrence included additional motor operated valve inspections, Environmental Qualification Program modifications, maintenance procedure revisions,

and additional training. These corrective actions resulted in the satisfactory resolution of the NRC concerns.

C.5.4 NRC Generic Letter 96-06

Based on performance observations of a containment penetration motor-operated valve at BVPS 1, DLC performed an evaluation of containment penetrations at both BVPS 1 and BVPS 2. The review determined that some BVPS 1 and BVPS 2 liquid filled containment penetration lines were not designed to allow for the effects of liquid filled thermal expansion during a Design Basis Accident. The corrective actions implemented included the addition of relief valves and system administrative controls. Based on the issues identified by BVPS and other industry events, the NRC issued Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Bases Accident Conditions." The Generic Letter addressed unanalyzed water hammer events and containment penetration over-pressurization as a result of containment ambient conditions during a Design Basis Accident. DLC performed an additional evaluation of the affected containment penetrations and heat exchanger components and concluded no further corrective actions were necessary. Design Basis Documents are being developed for the containment penetrations to organize and collate design bases information.

C.5.5 Seismic Walkdowns / I.E.B. 79-14

Based on Architect Engineer errors in analyzing seismically qualified piping at BVPS 1, the NRC issued I. E. Bulletin 79-14. DLC developed an inspection program to verify that the safety-related piping and support analysis agreed with the as-built construction of BVPS 1. Field verification walkdowns were performed on piping two and one-half inches and greater in diameter, correlating piping configuration with the as-built drawings and stress calculations. Modifications were completed such that the BVPS 1 piping configuration within the scope of the program was consistent with the design bases.

Based on the requirements of NRC Generic Letter 87-02, Unresolved Safety Issue A-46, BVPS 1 initiated a seismic review process based on the guidelines developed by the Seismic Qualification Utility Group. This review included equipment selection, field walkdowns, anchorage analysis, and separate reviews of relays, tanks, heat exchangers, and cable and conduit raceways. Identified deficiencies are being addressed in accordance with the Seismic Qualification Utility Group program requirements. This Generic Letter was not applicable to BVPS 2 due to the licensing time frame. Seismic SSC qualification was included in the initial licensing review of BVPS 2.

C.5.6 Appendix R and Fire Protection

For BVPS 1, DLC performed an Appendix R Fire Protection review in 1982 to ensure that the requirements of 10 CFR 50.48 were met. The review documented an analysis of safe shutdown capability based on the requirements of 10 CFR 50.48.

Field walkdowns were conducted at BVPS 1 to identify and confirm the location of raceways, cables, fire areas, protective equipment and combustible materials. Modifications were initiated as appropriate to meet the regulations. These modifications included cable routing changes, Diesel Generator control circuit changes, installation of a Dedicated Auxiliary Feedwater pump, and various fire protection upgrades.

A similar review was performed at BVPS 2 as part of the initial licensing to address the requirements of 10 CFR 50.48. The BVPS 2 Fire Protection Report was submitted as part of the original docketing in March 1984 and the Safe Shutdown Report was submitted in March 1987. Field walkdowns were also conducted at BVPS 2 prior to the Unit's start-up. These reviews and resulting modifications provide additional assurance that the BVPS Units are configured consistent with the design bases.

C.5.7 Spent Fuel Pool Practices / Information Notice 95-54

As a result of NRC Information Notice 95-54, which discussed industry fuel pool off-loading practices, DLC performed an evaluation which concluded that for the issues identified, the BVPS refueling practices were in compliance with the design bases requirements. NRC also evaluated the BVPS 1 and BVPS 2 UFSARs to determine whether the Units' off-loading practices were in compliance with the Licensing Basis as described in the UFSARs. The NRC report was consistent with the analysis performed by DLC in that it concluded that both BVPS 1 and BVPS 2 allowed for full core off-loads to be performed, and that the spent fuel pool cooling systems have adequate cooling capacity to accommodate such practices.

C.6 INDUSTRY EXPERIENCE

As noted above, BVPS has implemented verification and configuration controls designed to assure that SSCs are operated and maintained within the BVPS design bases. As an added measure, DLC also researches, reviews and analyzes relevant industry experiences and issues. This information is available to DLC in a variety of forms. Examples include:

- NRC generic communications (e.g., Bulletins, Information Notices)
- Vendor-supplied engineering and technical information and changes thereto (e.g., vendor documentation, Vendor Technical Information and 10 CFR 21 notifications)
- Equipment qualification data provided by the equipment vendor or qualification lab
- Industry-developed information (e.g., Significant Event Reports (SERs), Significant Operating Event Reports (SOERs), Nuclear Plant Reliability Data System (NPRDS))

Administrative procedures provide guidance for the evaluation and disposition of identified industry events and issues. The information is reviewed for applicability to

BVPS. Appropriate corrective actions are taken and programs, procedures and processes are modified to address industry issues and concerns.

C.7 QUALITY SERVICES UNIT

A review of Quality Services Unit oversight activities was conducted for the period of 1991 through November, 1996. During this period, Quality Services Unit activities included the review of appropriate processes and the evaluation of the effectiveness of those processes to maintain the BVPS SSC configuration and performance consistent with applicable design bases. These activities were conducted in accordance with the applicable Quality Services Unit established procedures. Quality Services Unit oversight activities have determined that even though some deficiencies, primarily in program implementation, have been identified, overall, the programs reviewed were found to provide additional assurance that BVPS SSC configuration and performance is consistent with design bases.

C.8 CONCLUSION

From the initial design, construction and start-up testing of the SSCs to their operation and modification in accordance with a comprehensive set of controlled programs, procedures and processes, the SSCs have been operated, reviewed and evaluated continually, and found to be consistent with their design bases. As these programs, reviews and initiatives have identified deficiencies, corrective actions have been or are in the process of being implemented, thereby improving the consistency of the SSCs with their design bases and diminishing the likelihood of the recurrence of comparable deviations. Neither the number nor character of those deviations was significant enough to be inconsistent with a finding of reasonable assurance that SSC configuration and performance is consistent with the design bases.

For the reasons discussed above, DLC has reasonable assurance that the BVPS 1 and BVPS 2 SSCs are configured and perform consistent with the design bases.

D. RESPONSE TO INFORMATION REQUEST (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

D.1 HISTORICAL OVERVIEW OF CORRECTIVE ACTION PROGRAM

10 CFR 50, Appendix B, Criterion XVI (Corrective Action) requires that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Since the licensing of BVPS 1 in 1976, a corrective action program has been in place, in accordance with 10 CFR 50, Appendix B, to identify, resolve and prevent the recurrence of problems that could degrade the quality of plant operations and safety. That corrective action program became applicable to BVPS 2 upon its licensing in 1987.

The BVPS corrective action program has been periodically evaluated for effectiveness through industry initiatives, internal and external reviews, audits and inspections. These evaluations resulted in corrective action program improvements. These improvements during the evolution of the corrective action program have strengthened DLC's ability to identify and trend problems.

The primary areas of focus of the initial BVPS problem identification and corrective action program were high level equipment failures, plant trips and NRC reportable events. The problems identified included design bases issues. The initial program had an operations focus and did not capture lower-level human performance events in other station groups.

The BVPS corrective action program remained essentially the same until 1993, when the "Incident Report" system was replaced by a "Problem Report" process for identifying and investigating conditions adverse to quality. As part of the Problem Report process, the threshold for documenting problems was lowered, in response to industry feedback and internal and external assessments which indicated the need for such a change. This broadening of the initial program's focus, to include a wider range of conditions and issues of lesser significance, reflected the increasing maturity of BVPS and the nuclear industry. The identification and resolution of design bases issues remained a major area of focus for the corrective action program.

Other BVPS processes have also identified actual or potential problems that required corrective action to prevent recurrence. Examples include self-assessments, vertical slice reviews, Quality Assurance audits, surveillances and inspections, industry reviews and NRC inspections. The Quality Assurance audits, surveillances, inspections, examinations

and assessments were conducted by the Quality Services Unit in accordance with the requirements of 10 CFR 50, Appendix B. Quality Assurance discrepancies identified were independently documented, tracked, trended and corrective actions were verified to be completed by the Quality Services Unit.

DLC implemented a management initiative to improve the implementation and effectiveness of the Problem Report process. One of these measures was to transfer responsibility for the Problem Reporting program to the Nuclear Safety Department. BVPS management also provided additional resources that were needed to strengthen the implementation of the corrective action program and to address areas of concern.

October 1996 Quality Assurance Audit

In October 1996, the DLC Quality Services Unit audited the implementation and effectiveness of the BVPS corrective action program. This audit identified significant concerns and determined that the program was not fully effective. Based on this audit, DLC implemented additional enhancements to improve performance in several areas, such as effectively identifying problems at yet a lower threshold, accurately determining the causes and the extent of those problems, taking action to correct problems, and preventing recurrence.

Another programmatic improvement was the implementation of the "Condition Report" process on January 1, 1997, to replace the Problem Report process. In an effort to resolve issues noted in the October 1996 Quality Services Unit audit, the Condition Report process instituted a more precise system of problem classification. The problem classification system was designed to ensure that the review of a problem and associated actions taken to prevent recurrence are proportionate to the significance of the problem. Notwithstanding the deficiencies identified in the Quality Assurance audit, design bases issues were being identified through the Problem Report process.

FPI Assessment Report

As part of a management initiative to resolve identified issues, DLC requested Failure Prevention International Inc. (FPI) to review Problem Reports and programs for design and configuration control at BVPS. FPI was also requested to provide conclusions and recommendations for improvements in these areas.

During October and November 1996, FPI conducted a common cause analysis of the design and configuration control related issues identified by the Problem Report process. FPI issued the final report on January 15, 1997. It concluded that the Problem Report process had too high a threshold for problem identification and that problem investigation was weak. The report also identified potential weaknesses with the configuration management process and questioned DLC's capabilities to effectively deal with the anticipated increase in the volume of problems that were expected to be identified as a result of ongoing improvements in the corrective action program and action to heighten staff awareness of non-conforming conditions.

At the time of this Response, DLC had preliminarily evaluated the FPI conclusions and recommendations. The new Condition Report program implemented on January 1, 1997 included provisions which address many of the issues in the FPI report. The remaining issues are being reviewed and considered by BVPS management.

DLC management has also strengthened its commitment to encourage employees to identify and report problems. In addition, DLC management is attempting to facilitate problem identification by broadening the scope of conditions covered by the problem reporting system. After implementation of these management initiatives, the number of problem reports has significantly increased. For example, approximately 200 problem reports were generated in 1992. The number has increased each year since, with over 1,800 problem reports generated in 1996.

D.2 PROCESS FOR IDENTIFYING AND CORRECTING PROBLEMS

D.2.1 Overview of Condition Report Program

The problem identification and Condition Reporting Program described below reflects the program currently in effect at BVPS. In most ways, the process is very similar to the Problem Report process previously used. The most significant differences are identified in the description that follows.

The requirements of the current Condition Reporting Program are delineated in two (2) procedures: initiation of Condition Reports and processing of Condition Reports. The use of separate procedures for initiating and processing a Condition Report is intended to make it easy for site personnel to identify and document conditions adverse to quality while providing separate detailed instructions for administering the Condition Report Program. Condition Report processing includes administration, screening for operability and reportability, investigations (including actions to determine the extent of the adverse condition), identification of cause, identification of corrective actions to prevent recurrence, implementation and verification of corrective actions, trending and analysis of data, independent closure of the Condition Report and the performance of effectiveness reviews.

D.2.2 Problem Identification and Documentation

BVPS personnel (including contractor personnel) are responsible for promptly identifying and documenting potential adverse conditions that relate to the design, maintenance or operation of the plant. The Condition Report is the mechanism used at BVPS to identify and document these conditions.[5] (Previously problems were documented in Problem

---

[5]Deficient conditions involving plant hardware that requires routine maintenance, excluding rework items, are documented through the Maintenance Work Request process, rather than via a Condition Report. See Section C for further discussion of the Maintenance Work Request process. Maintenance rework items are documented with Condition Reports.

Reports.) As a conservative measure, BVPS personnel are directed by procedure to generate a Condition Report when in doubt about how to address an issue.

The Condition Report process is designed to facilitate the prompt identification and correction of problems. BVPS employees receive training on the identification and documentation of conditions adverse to quality, including those conditions related to plant design bases and configuration management, through General Employee Training and General Employee Refresher Training. Selected individuals, closely associated with the corrective action program, received formal classroom training on the Condition Report process. In addition, DLC has distributed site-wide a brochure explaining the new Condition Report process and the responsibility of BVPS personnel to initiate a Condition Report to document actual or potential adverse conditions, including those conditions that involve design bases and configuration management issues.

Individuals who discover conditions that may have an immediate impact on safety or operability, involves a fire, a significant personnel-related industrial safety accident, an uncontrolled release of radioactive material, a threat to plant security, or may otherwise be reportable to an outside agency, are required to notify the control room immediately prior to initiating a Condition Report.

In a non-emergency situation, the individual who detects the problem promptly originates a Condition Report describing the nature of the potential condition or problem, the cause (if known) and any immediate corrective action taken. To assist employees in problem identification, the Condition Report initiation procedure provides examples of the types of conditions that should be documented. Some examples of possible problems related to design bases and configuration management include:

- Violation of a procedural requirement
- Abnormal damage, degradation or failure of equipment
- Unplanned, unexpected, or unanalyzed operating events or conditions
- Deficiencies in the design of SSCs that affect nuclear safety
- Documentation discrepancies
- Deviations from requirements in the Technical Specifications and Fire Protection Report
- Significant discrepancies discovered during plant surveillances or procedure performance
- Deviations from requirements in the UFSAR
- Deviations from design documents
- Discovery that a previously installed item does not meet specified installation requirements
- Adverse trends in equipment or system performance

- Deviations from requirements in the ASME Code
- Failure of independent verifications to identify discrepancies

Other BVPS processes may also identify conditions that require corrective action. The Condition Report process was designed to collect the problems and conditions from these other processes to ensure that corrective actions to prevent recurrence are identified, tracked and trended in a consistent manner. For example, Quality Assurance discrepancies, identified by the Quality Services Unit, and NRC inspection issues (violations, unresolved items, inspector follow-up items) should be documented by using a Condition Report. The input of corrective actions from other processes into the Condition Report system provides centralized tracking and trending of problems, issues and conditions. The Condition Report centralized tracking and trending is a new feature of the corrective action program.

The Condition Report process now identifies, tracks and trends Quality Assurance audit, surveillance and inspection discrepancies generated by the Quality Services Unit. To preserve Quality Services Unit independence, special provisions for Quality Services Unit initiated condition reports are included in the Condition Report process. When the Quality Services Unit initiates a Condition Report as a result of its oversight responsibilities, it identifies the classification, due date and the organization responsible for investigation and resolution. The Quality Services Unit also approves the corrective action response and verifies completion of corrective actions for the Condition Reports it initiates.

After a Condition Report is initiated, the supervisor of the originator of the Condition Report reviews the Condition Report for completeness. The supervisor also records the "quality barrier" level at which the condition was identified. The four (4) quality barrier levels, in order of desirability, are: Individual/Work Group, Supervision/Management, Internal Oversight and External Oversight. DLC management emphasizes a questioning attitude to encourage self-identification of problems at the work activity level. A proactive attitude in the work force is encouraged to promote attention to and identification of problems at the lowest possible "quality barrier." The quality barrier level determination is collected so that it can be later trended to determine whether management expectations for identifying problems at the lower barrier levels are being met. After the supervisory review, the Condition Report is hand-delivered to the control room.

D.2.3 Operability and Reportability Review

In the control room, the Nuclear Shift Supervisor, a licensed Senior Reactor Operator, reviews the Condition Report to identify immediate impacts on operability or reportability and decides whether to notify senior plant management. NRC Generic Letter 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Non-conforming Conditions and on Operability," is used to assist in operability determinations. The Nuclear Shift Supervisor

decides whether a Basis for Continued Operation is required. If an event is determined to be reportable, the Nuclear Shift Supervisor notifies the NRC and/or other outside agencies. When appropriate, the Nuclear Shift Supervisor also initiates an immediate preliminary investigation and critique of an incident.

Shift Technical Advisors in the control room review newly generated Maintenance Work Requests for safety-significant issues, including design bases-related issues. The Nuclear Shift Supervisor is notified of any safety significant issues identified by the Shift Technical Advisor review for determination of operability and reportability. This is designed to ensure that safety significant issues documented in a Maintenance Work Request do not bypass the Condition Report process. System engineers and other site personnel assigned responsibility for a particular system also review Maintenance Work Requests for Condition Report applicability.

10 CFR 50.72 and 10 CFR 20.2202

The Nuclear Shift Supervisor is required to notify the NRC of the declaration of any one of the Emergency Classes specified in the Emergency Plan or any other conditions specified in 10 CFR 50.72 and 10 CFR 20.2202, including conditions outside the design bases of the plant. Procedures instruct personnel to conservatively report questionable events. The control room staff uses the guidance provided by NUREG 1022, to assist in making reportability determinations.

10 CFR 50.73

When a specific event is determined to be reportable, a Licensee Event Report is written by the Nuclear Safety Department, reviewed by the OSC and NSRB, approved by the Plant Manager and sent to the NRC within 30 days of the event. Conditions requiring a Licensee Event Report, excluding special reports, should be documented by initiating a Condition Report.

10 CFR Part 21

If, during the investigation of a Condition Report, information is obtained that reasonably indicates a basic component supplied for use at BVPS could compromise safety, or if an observation is made of a facility activity that could compromise safety, a '10 CFR 21 Decision Checklist' is prepared. The Nuclear Safety Department is notified. After identification of the potential defect or nonconformance, the Nuclear Safety Department determines whether the issue is reportable in accordance with 10 CFR 21. A 10 CFR Part 21 notice, which is determined to be applicable to BVPS, is an example of a condition that should be documented by a Condition Report.

Basis for Continued Operation

BVPS endorses the use of Generic Letter 91-18. The Nuclear Shift Supervisor makes an operability determination upon discovery of degraded conditions of equipment where

performance is called into question, discovery of nonconforming conditions where the qualification of equipment (such as conformance to codes and standards) is called into question, or discovery of an existing but previously unanalyzed condition or accident. If a definite operability status can not be established, the Nuclear Shift Supervisor will decide if reasonable expectation exists that the SSC will perform its specified function, based on the best available information. When the Nuclear Shift Supervisor determines that the SSC is operable by reasonable expectation, the Nuclear Shift Supervisor will notify the System and Performance Engineering Department, to begin and coordinate a Basis for Continued Operation Evaluation to review and document the rationale for continued operations.

A Basis for Continued Operation Determination exists once agreement on a duration time for the Basis for Continued Operation, commensurate with the safety significance of the condition, has been reached and the Basis for Continued Operation Evaluation has been approved. A Basis for Continued Operation Determination is not intended to be used to justify long term plant operation with a degraded or nonconforming SSC since an unintended deviation from design of an SSC described in the UFSAR could be interpreted as a de facto design change to the facility.

D.2.4 Problem Classification and Investigation

Following control room review, Condition Reports are classified by the Condition Report Program Administrator into one (1) of five (5) categories based on safety significance and other considerations. The previous Problem Report process included only two (2) classification categories. The Condition Reporting system thus provides for improved prioritization and allocation of resources based on a more refined determination of the relative significance of a problem.

The Condition Report procedure provides a detailed definition of each problem category including examples. Category 1 conditions are deemed to have the highest safety and/or operational significance and therefore require the highest level of management review and technical analysis. Category 5 conditions are of the lowest significance and are identified for the purpose of trending. Category 5 conditions are generally well understood based on past operating experience, with related corrective actions either complete or in progress.

The time available for the investigation of a Condition Report is based on the Condition Report Program Administrator's classification of the problem. For example, for Category 1 and Category 2 conditions, the investigation and identification of corrective action(s) must be completed within 14 days and 21 days for reportable and non-reportable issues, respectively. Any extension of the period for investigation requires the approval of the Condition Report Program Administrator. Subsequent extensions require the approval of a DLC Nuclear Power Division Vice President.

Following classification of the conditions, the Condition Report is assigned to a responsible organization for investigation. The assignment of responsibility for

individual Condition Reports is done at the Managers' meetings, which are held daily during the normal work week. This discussion of Condition Reports and other plant activities facilitates the early identification and effective resolution of emergent issues. DLC management's expectations regarding involvement in assuring timely and effective resolution of corrective actions have been, and continue to be, communicated during these meetings.

D.2.5 Extent of Problem Determination

For Category 1, Category 2 and Category 3 conditions, an "investigation checklist" is required to be completed. Some of the issues considered on this checklist include training, management's expectations, Maintenance Rule impact, self-checking and effect on the opposite Unit. The "investigation checklist" is designed to ensure that a variety of generic issues are considered to assist in determining the cause and implications of the condition.

Next, an "extent of condition determination" identifies related operating experience at BVPS and/or at other nuclear power plants. This includes a review of the Nuclear Network, OERs, Safety Evaluation Reports, SOERs, Licensee Event Reports, NRC violations and the corrective actions taken in response to past occurrences. The "extent of condition determination" also considers potential similar or generic applications and/or common failure characteristics (e.g., personnel error, procedural deficiency, environmental conditions). Based on the significance of the problem, a multi-discipline Event Response Team may be established to perform the Condition Report investigation.

D.2.6 Cause Analysis to Identify Actions to Prevent Recurrence

After the "extent of condition determination" is completed, a "cause analysis" is performed to make a final determination of the cause(s) of the condition and to identify the appropriate corrective action(s) to prevent recurrence. A formal root cause analysis is performed for Category 1, 2 and, in some cases, Category 3 Condition Reports. An apparent cause determination is made for Category 3 Condition Reports that do not receive a formal root cause analysis. Category 4 conditions, by definition, have low nuclear safety, regulatory, operability or power production impact. The evaluation of a Category 4 event is limited to the identification of the apparent cause and remedial actions to correct the condition.

Previously, BVPS performed root cause analyses using a system based on the INPO Human Performance Evaluation System. In 1993, a self-assessment of BVPS's implementation of the Human Performance Evaluation System root cause analysis methodology identified deficiencies in that system. To address these deficiencies, a new proprietary root cause analysis system, called TapRoot™, was implemented in late 1994. In late 1995, a Quality Services Unit review of the implementation of the TapRoot™ process was conducted. This self-assessment concluded that the TapRoot™ process was an improvement over previous site methods of root cause analysis. However, some areas

for improvement were identified. For example, the Quality Services Unit noted a lack of procedural guidance, leading to inconsistency of TapRoot™ reports. The review also found that TapRoot™ recommendations were not always tracked, trended or followed up, and that no procedural requirements for performing "extent of condition reviews" existed. The current Condition Report process has been enhanced to address these areas.

A site administrative procedure on root cause analysis now provides a consistent site-wide method of applying formal root cause analysis techniques. A trained individual not directly involved with the event performs each formal root cause analysis. The site administrative procedure and associated training provide for a common language and approach to root cause evaluations. They are also designed to ensure that the root causes and appropriate corrective actions to prevent recurrence of an undesired event or adverse trend are identified.

Tracking, trending and verification of corrective actions resulting from TapRoot™ evaluations are now part of the Condition Report process. In addition, if TapRoot™ identifies a cause for an event for which no corrective action is recommended, then justification is required. As previously discussed, the Condition Report process includes formal "extent of condition reviews."

Recent reviews of the TapRoot™ process have been positive. One example is the NRC's recent assessment of the root cause analysis of the reactor coolant pump high bearing oil reservoir condition. This analysis was characterized as thorough and detailed.

D.2.7 Corrective Action Implementation

As stated previously, corrective actions are identified as part of the investigation of a Condition Report. At least one (1) corrective action is developed for each cause identified by the cause analysis. A corrective action to prevent recurrence is also developed for each cause identified during the extent of condition determination. Corrective action processes for design bases issues include: Design Change Packages, Technical Evaluation Reports and Temporary Modifications. (See Section A for further discussion of these processes.) The corrective action is the end product of such a process. For example, a corrective action to prevent recurrence would be to implement a design change rather than issue a Design Change Package.

Following completion of the investigation of a problem, a corrective action implementation schedule is established. Different management levels (up to and including the Senior Vice President, Nuclear Power Division are required to approve the corrective action depending on the duration of the implementation schedule. (The greater the duration, the higher the level of management approval required.) This approval process is designed to reinforce management's expectations for prompt implementation of corrective actions. Condition Reports and their associated corrective actions are entered into a computer data base for tracking and trending.

Upon completion of the required corrective action, a Corrective Action Response Form is completed by the assigned department and returned to the Condition Report Program Administrator. The Condition Report Program Administrator then coordinates a review of Category 1 and Category 2 Condition Reports by the NSRB. The close-out of Category 1, Category 2 and Quality Services Unit Condition Reports is also independently verified by the assigned organization. The Quality Services Unit also performs its own verification of Quality Services Unit initiated Condition Reports. The Condition Report Program staff updates the computer database and assembles the documents pertaining to the investigation and corrective actions for transmittal to Beaver Valley Records Center prior to formally closing out the Condition Report.

Category 3 and Category 4 Condition Reports may be closed out prior to the actual completion of corrective actions, provided that the corrective actions have been initiated and are being tracked as part of another auditable, controlled and proceduralized process. Examples of such processes are Maintenance Work Requests, Temporary Modifications and Design Changes.

D.2.8 Effectiveness Reviews

The Condition Report Program Administrator, responsible organization managers, or the Nuclear Safety Review Board determine the need to conduct an effectiveness review of corrective actions that have been implemented. The effectiveness review verifies that the problem or condition has not recurred and thus each corrective action has prevented recurrence of its associated cause(s). If the corrective actions are determined to have been ineffective, a new Condition Report is initiated and responsibility for resolution is assigned to a higher level of management. Category 1 and Category 2 Condition Reports typically receive an effectiveness review 90 to 120 days following the completion of corrective actions. DLC will conduct an effectiveness review of the Condition Report process in 1997.

D.2.9 Trending

The Condition Report Program Administrator is responsible for establishing appropriate performance indicators associated with each condition and its related cause(s). Each Corrective Action is assigned an activity code to characterize the activity being performed for each cause, a cause code to address each cause and an organization code to indicate the personnel responsible for resolution. A trend analysis report on Condition Report data is required to be prepared and updated at least quarterly. Identified adverse trends require the initiation of a new Condition Report.

As discussed above, the identification and trending of the quality barrier level at which Condition Reports are initiated will show whether problems are properly identified and reported at the lowest possible level.

D.3 EMPLOYEE CONCERNS RESOLUTION PROGRAM

The Employee Concerns Resolution Program provides an alternative, confidential and impartial channel for communication of employee, contractor and consultant concerns or issues for investigation and resolution. The Ombudsman position serves as a neutral facilitator to resolve issues that cannot be resolved through normal management channels in order to encourage personnel to identify their concerns without fear of retribution. The Ombudsman is specifically available to listen to concerns in the area of Nuclear Safety, Quality, or other matters that could result in a potential violation of laws or regulations. Concerns that are communicated to the Ombudsman are confidential. The Ombudsman may share the problem or concern with the Senior Vice President, Nuclear Power Division as needed without identifying the source. Where appropriate, Condition Reports may be initiated to identify and implement corrective actions for the concerns communicated to the Ombudsman.

D.4 SSFE PROGRAM CORRECTIVE ACTION PROCESS

The SSFE process incorporates the basic elements of a comprehensive corrective action program. It has implemented a structured process that identified issues, prioritized them in accordance with their safety significance, established completion schedules and tracked issues to completion.

For the SSFEs conducted from 1988 to 1991, the comments generated were categorized in descending order of safety significance as either a finding, observation or recommendation. Categorization was established by the SFFE team by using the SSFE manual to guide its review and judgment. Categorization of SSFE items was also reviewed by the SSFE Project Manager, the SSFE Management Overview Committee, the Independent Safety Evaluation Group, and the Operations Experience Department. Each observation was assigned to an appropriate department for resolution.

Once assigned, the responsible department was required to provide either closure or an acceptable approach for closure along with scheduled corrective actions. Responses to observations were modeled on the format used by the Quality Services Unit (i.e., a summary of corrective action taken/planned including schedules and action taken to prevent recurrence).

The remedial actions taken for SSFE observations have been documented in individual observation files. Progress on each observation was initially tracked utilizing a separate database. Long-standing items were later transferred to the site commitment tracking system for additional visibility. The status of remedial actions taken for SSFE observations was periodically reviewed by the SSFE Management Overview Committee.

The closure of observation items required evidence that the corrective actions were completed. Acceptable evidence included SSFE team cross check/verification of stated activities and a letter of closure. Formal closure of observation items was also documented and has been maintained in the individual SSFE files.

For these reasons, the SSFE process can be viewed as a corrective action process that has been specifically tailored to address, among other issues, the consistency between the design bases of the Unit's SSCs and related procedures for operating, maintenance and testing.

E. RESPONSE TO INFORMATION REQUEST (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

---

E.1 OVERALL EFFECTIVENESS

DLC concludes that its current programs and processes are effective overall in providing reasonable assurance that the configuration of BVPS is maintained consistent with its design bases for the following reasons:

- Comprehensive and controlled engineering design and configuration control processes are implemented in accordance with 10 CFR 50, Appendix B,
- Design bases requirements are translated into appropriate operating, maintenance and testing procedures,
- Programs, procedures and processes are implemented to maintain the configuration of SSCs and their performance consistent with the design bases,
- A corrective action process is implemented that includes the identification of problems, implementation of corrective actions, including actions to determine the extent of problems, actions to prevent recurrence of problems, and reporting to the NRC,
- Ongoing routine and special initiative assessments to evaluate the effectiveness of the programs and processes are conducted,
- Routine internal oversight programs and assessments are conducted,
- External oversight assessments, reviews and inspections are conducted.

When deficiencies are identified, corrective actions are processed, and when appropriate, programs are modified to preclude their recurrence.

The above programs and processes, combined with DLC management's clear communication of expectations that the plant configuration be consistent with the design bases, implementation of a continuing critical self-checking process, results of assessments conducted to date and analyses of recent plant events provide reasonable assurance that the configuration of BVPS 1 and BVPS 2 is consistent with the design bases.

E.2 EFFECTIVNESS OF ENGINEERING DESIGN AND CONFIGURATION CONTROL PROCESSES

As discussed in Section A of this Response, DLC has implemented programs, processes and procedures for engineering design and configuration control which provide additional confidence that each of BVPS 1 and BVPS 2 is configured and operated consistent with its design bases. These comprehensive programs, procedures and processes cover the scope of activities delineated in the applicable standards. Several program elements provide internal support for the effectiveness of these programs. Among these program elements are interfaces between departments that have relevant expertise for a proposed activity, independent design reviews, and multi-level and multi-disciplinary reviews which help to assure that the appropriate technical and management areas of expertise are brought to bear on a proposed activity.

DLC's engineering design and configuration control processes and the outputs of these processes have been reviewed extensively, both internally and externally. Assessment results generally demonstrate that design and configuration control processes have been effective. As expected with any effective Quality Assurance Program, deficiencies have been identified. Deficiencies were prioritized and corrective actions have been or are in the process of being implemented. These deficiencies have not called into question the reasonable assurance that BVPS 1 and BVPS 2 are configured and operated consistent with their design bases.

E.3 EFFECTIVENESS OF PROCESSES FOR TRANSLATING DESIGN BASES REQUIREMENTS INTO OPERATING, MAINTENANCE, AND TESTING PROCEDURES

As discussed in Section B of this Response, effective processes are in place to translate design bases requirements into procedures. The processes which control the development of and revisions to operating, maintenance and testing procedures are designed to ensure that design bases requirements are appropriately incorporated into these procedures by providing for the incorporation of source requirements including design bases requirements, multiple levels of review, walkdowns where appropriate, and controlled distribution of the procedures. Appropriate personnel have been trained in these procedures. DLC policies require personnel to use controlled procedures, and to initiate procedure changes when needed. Management has clearly communicated its expectations in this regard.

Effectiveness of the translation of design bases information has been assessed both internally and externally. Of special note are the results of the vertical slice reviews. As discussed in Section B, the procedures accurately reflected design bases information. Of the deviations that were found in the translation of design bases information into procedures, only a few were significant enough to be characterized as findings. Other deviations were of lesser significance. Corrective actions have been or are in the process

of being implemented. Additional vertical slice-type assessments will be conducted and corrective actions will be implemented for identified deviations. These circumstances support the conclusion that there is reasonable assurance that design bases information has been accurately translated into operating, maintenance and testing procedures.

E.4 EFFECTIVENESS OF CONFIGURATION MANAGEMENT PROCESS

As discussed in Section C of this Response, processes are in place to ensure that SSC configurations and performance are consistent with the design bases. DLC implements a Configuration Management Program which is designed to ensure that the plant configuration and configuration changes remain within the design bases. In addition, DLC has implemented programs and processes, such as system configuration and work controls, to provide additional assurance that work activities maintain the SSC configuration and performance consistent with its design bases.

In addition, the results of routine plant testing programs, including surveillance tests, demonstrate that systems can perform as required to meet design bases requirements. Where deviations are discovered, corrective actions have been or are in the process of being implemented. Years of plant operation using operating procedures that incorporate design bases requirements and continuing successful surveillance testing provide additional confidence that design bases requirements are being met.

Routine walkdowns by system engineers, implementation of the Maintenance Rule, special walkdowns and configuration checks for specific purposes such as seismic design and NRC I. E. Bulletin 79-14 requirements have contributed to maintaining appropriate system configuration.

Effectiveness of the configuration management process in maintaining consistency between the SSCs and their design bases has been assessed both internally and externally. Of special note are the vertical slice reviews. As discussed in Section C, the SSCs were found to be consistent with their design bases. Of the differences between SSCs as configured and their design bases, only a few were significant enough to be findings. The other deviations were of lesser significance. Corrective actions have been taken or initiated for the differences discovered. Additional vertical slice-type assessments will be conducted and corrective actions will be implemented for identified deficiencies. These circumstances support the conclusion that there is reasonable assurance that the configuration and performance of the BVPS 1 and BVPS 2 SSCs are consistent with their design bases.

E.5 EFFECTIVNESS OF THE CORRECTIVE ACTION PROGRAM

DLC management continues to actively encourage employees to identify and report problems. In furtherance of this expectation, DLC management has made efforts to lower the site threshold for problem reporting and has recently modified the corrective action process to accommodate the resulting increased number of issues reported. An additional

effort to identify design bases issues has been made in this area by conduct of the vertical slice reviews, as discussed in Section D.

Although problems were being reported and corrected under the prior corrective action program, and some strengths in the previous program were noted by the NRC, DLC recently implemented significant enhancements to the Problem Report program in response to deficiencies recognized by BVPS management. A Quality Services Unit audit also identified deficiencies in that program. The recently implemented Condition Report process is an integral part of the enhanced corrective action program. Effective implementation of this Condition Report process will provide additional assurance that as conditions adverse to quality are identified, they will be appropriately dispositioned, programs will be modified accordingly and reporting to the NRC will be performed as required.

E.6 EFFECTIVENESS CORROBORATION THROUGH ASSESSMENTS AND OVERSIGHT

DLC has implemented assessment processes that review and evaluate the design and configuration control programs and processes. These assessment and oversight activities are in addition to the review and approval processes that are incorporated as integral parts of the design bases control processes. As discussed in the previous Sections of this Response, numerous reviews, assessments, audits, and oversight of the DLC design and configuration control processes and programs have been and continue to be conducted by plant personnel, third parties, and the NRC. Included are routine internal assessments, special internal assessments, routine internal oversight, and external oversight. The results of these reviews, assessments, audits, and oversight activities provide added confidence in the effectiveness of the processes that have been implemented.

E.6.1 Self-Assessment

DLC has implemented a site-wide self-assessment program. Self-assessments can be initiated to identify enhancements intended to improve general performance and efficiency. Self-assessments may also be performed as a result of the identification of recurring NRC findings or Quality Services Unit deficiencies, Condition Reports, or adverse trends. The Department Manager establishes the scope of the assessment and selects the review team personnel.

The self-assessment program includes evaluations of design and configuration control activities. For example, in 1996, the Nuclear Engineering Department conducted self-assessments of the revised Design Change Package and Technical Evaluation Report processes. The purpose of the assessments was to determine the effectiveness of these new processes. They concluded that the new processes were effective; however, they also recommended several programmatic improvements.

Quality Services Unit audits of the self-assessment program have identified recommended enhancements to the program including improvement in self-assessment

topic selection, response, and corrective actions. Corrective actions have been implemented and a follow-up effectiveness review is scheduled to be completed in 1997.

Engineering Assurance is another routine self-assessment process. The Nuclear Engineering Department established the Engineering Assurance Section to evaluate design and configuration control activities in order to improve the overall quality of engineering processes and products. Engineering Assurance conducts independent reviews, evaluations, assessments and analyses of Nuclear Engineering Department technical and administrative activities, including Problem and Condition Reports, independent of Quality Services Unit assessments and audits. The Engineering Assurance reviews provide critical assessments of management style, organization, and a review of programs in selected areas for the identification of programmatic issues. The Engineering Assurance Section advises the Nuclear Engineering Department Manager of any noncompliances identified as a result of their reviews.

The NRC noted that the establishment of Engineering Assurance is considered to be a good initiative with a positive impact on the quality of engineering activities. Nevertheless, a recent internal review found that there were additional opportunities for improvement. These recommendations are being evaluated and considered.

E.6.2 Quality Services Unit

Quality Services Unit (QSU) audit, surveillance and assessment activities independently evaluate design and configuration control programs, processes and activities. QSU oversight processes include the review of major program activities related to design and configuration control such as design control, configuration control, document control, the Maintenance Rule program, UFSAR and Technical Specification revisions, 10 CFR 50.59 evaluations and other related programs, processes and activities.

QSU regularly assesses the scope, adequacy, and compliance of the design control program with 10 CFR 50, Appendix B, and ANSI N45.2.11 in order to provide added confidence that the BVPS's SSCs will perform as designed. QSU fulfills this function through comprehensive audits, surveillances and assessments. Audits are performed at a frequency commensurate with their importance to safety, regulatory commitments and past performance results. Audits evaluate both compliance-based and performance attributes obtained from quality verification and objective evidence of quality achievement. For example, a QSU audit would determine not only whether a particular procedure was followed, but also whether the procedure accomplished its intended purpose.

QSU routinely supplements the site membership on audit teams with additional participants from other utilities to provide additional technical expertise relevant to that specific audit. This practice not only provides for greater technical expertise for the audit activities, but also provides an additional measure of independence as well. Consultant members of the ORC also participate as QSU audit team members.

In the development of audit checklists, QSU reviews and incorporates pertinent information including design bases information from the:

- UFSAR,
- Technical Specifications,
- NRC Generic Letters, Bulletins, and Information Notices,
- NRC Inspection Procedures and NRC Inspection Reports,
- Licensee Event Reports (LERs),
- Industry experience,
- Site experience,
- Site management input, and
- Site and department administrative and technical procedures.

In addition, the QSU Manager has issued a written direction to QSU personnel to include review of pertinent UFSAR information in audit preparation, to include pertinent UFSAR information in audit checklists, and to verify that implementing procedures and activities are conducted in accordance with the UFSAR.

DLC review of QSU oversight activities conducted since 1991 confirms appropriate evaluations of design and configuration programs and processes. For example, QSU periodic audits have been conducted on Operations and Technical Specifications, Maintenance, Technical Support, Engineering, Plant Configuration Control, Design Control, and the Corrective Action process.

DLC is also sensitive to activities that may inadvertently bypass or circumvent established design and configuration control processes. QSU oversight includes examination of Design Changes, plant modifications, Temporary Modifications, Maintenance Work Requests, infrequently performed tests, and other similar activities.

QSU audit results confirm that appropriate emphasis is being placed on the design and configuration control processes through these oversight activities. This oversight process also has identified deficiencies involving design and configuration control programs, principally in their implementation. Examples of deficiencies identified by QSU include failure to update certain information on the Master Equipment List, less than timely update of Technical Evaluation Report-related documentation, and less than effective implementation of the Fuse Control program. The deficiencies are tracked and corrective actions have been or are in the process of being implemented and, when appropriate, the programs modified to prevent recurrence. The overall result is that the design and configuration control processes, while acceptable, continue to be improved and enhanced, and continuing emphasis is being applied to effective program implementation.

E.6.3 Special Internal Assessments

As has been discussed in the previous Sections of this Response, DLC has conducted a number of special internal assessments of the design and configuration control processes. Vertical slice reviews have been conducted on ten (10) safety-related systems at BVPS 1 and one (1) safety-related system at BVPS 2 to verify that they are capable of performing their intended safety functions under normal operating and accident conditions.

The BVPS 1 systems were:

- Auxiliary Feedwater System (1988)
- Quench Spray System (1988)
- Emergency Diesel Generator (1988)
- Reactor Plant River Water System (1989)
- Recirculation Spray System (1989)
- Residual Heat Removal System (1989)
- Supplementary Leak Collection and Release System (1990)
- Electrical Distribution System (1991)
- River Water System (SWOPI) (1994)
- Safety Injection System (SSFI) (1995)

The BVPS 2 system was the Service Water System (SWOPI) (1994).

These vertical slice reviews are used to corroborate the effectiveness of the design control programs and show that the systems are capable of performing their intended safety functions under normal operating and accident conditions.

In November 1996, DLC commissioned two independent, outside experts to evaluate the vertical slice review process and previously conducted reviews. The independent reviewers concluded that the DLC SSFE process was a strong and worthwhile initiative that led to in-depth technical assessments of safety systems. The independent reviewers found that the SSFE teams were highly experienced and multi-disciplinary and were strengthened by the presence of representation from outside organizations, which provided additional independence and an outside perspective to the SSFEs. The independent reviewers concluded that the SSFEs were effective activities which identified findings and observations that resulted in upgrades to existing plant programs, documentation and procedures.

Similar conclusions have been drawn by the NRC. The NRC has included observations about the vertical slice reviews (SSFEs, SWOPI and SSFI) in several inspection reports. The bulk of those observations have been favorable. Two (2) early criticisms, regarding the promptness of corrective actions and a concern about aggressiveness of the SWOPI,

were satisfactorily resolved. Since then, the NRC has found the DLC's vertical slice reviews to be effective initiatives.

The independent reviewers identified enhancements for consideration in the conduct of future vertical slice-type assessments. These included revising the review procedures to reflect current industry and NRC guidance and the integration of review results into site corrective action, operability, reportability, and 10 CFR 50.59 programs.

The independent reviewers also identified weaknesses in the timeliness of closure of some observations and in the process for identification, evaluation, and disposition of generic and programmatic issues arising from the vertical slice reviews. In response, DLC entered the potential generic and programmatic issues which were identified in the previous SSFEs into the site commitment tracking system for disposition.

In late 1996, an independent, multi-disciplinary team conducted two (2) focused Vertical Slice Reviews for BVPS 2 on the Auxiliary Feedwater and Safety Injection Systems. These reviews were conducted to provide an additional independent assessment of the effective implementation of the design control program. The primary objective of these focused Vertical Slice Reviews was to assess continuing consistency with the Unit's design bases for the selected system through an in-depth review of plant information, engineering documents, programs and procedures. The secondary objective of the focused Vertical Slice Reviews was to identify programmatic deficiencies in order that follow-up actions could be taken to prevent recurrence.

Based on these reviews, it was concluded that the design bases of the BVPS 2 Auxiliary Feedwater and Safety Injection Systems have been adequately maintained. No issues affecting operability were identified. Principal design information for both the Auxiliary Feedwater and Safety Injection Systems were determined to be generally accurate and consistent. A number of discrepancies associated with changes such as updates to drawings indicated weakness in attention to detail at the time the changes were made. Condition Reports have been generated as appropriate for the most significant discrepancies. Corrective actions have been or are in the process of being implemented.

The focused Vertical Slice Reviews noted that over time there has been significant improvement in the formality, rigor, and documentation of activities that maintain the design bases and control the configuration of the facility.

Focused Design Reviews were undertaken at BVPS 1 after the Independent Safety Evaluation Group conducted a post-trip analysis of a BVPS 1 turbine/reactor trip on May 31, 1996 that was characterized as insightful by the NRC, and led to the discovery of a design deficiency with the BVPS 1 AMSAC. In response to this discovery, DLC took a broad approach to corrective actions by initiating a series of Focused Design Reviews. The purpose of these reviews was to evaluate the functional adequacy of the as-built design for the selected systems.

The following systems were selected because they represented generically engineered modifications that were customized for BVPS and installed by Design Change Packages: AMSAC, Inadequate Core Cooling Monitor/Reactor Vessel Level Indication, High-Energy Line Break Isolation and Post Accident Sampling Systems. Several weaknesses were found relating to functional testing, setpoint bases, and configuration control. A review showed that aside from the AMSAC situation, the remaining deficiencies were not safety significant. Accordingly, they did not reflect significant problems with the processes applied to the Design Change Packages. Corrective actions for the resolution of these weaknesses have been assigned and are being tracked in the site commitment tracking system. Upon their completion, the Design Change Package process will be more effective. For these reasons, DLC concludes that the special assessments support the overall effectiveness finding of its design control processes.

E.6.4 Routine Oversight

Multiple levels of oversight of the BVPS programs and processes help to reinforce the effectiveness of the conduct of activities that affect design and configuration control. This oversight includes QSU, as discussed in E.6.2, the OSC, the Independent Safety Evaluation Group, the Nuclear Safety Review Board, and the ORC.

E.6.4.1 Onsite Safety Committee

The OSC, through its recommendations to the General Manager, Nuclear Operations or his designee on matters related to nuclear safety, is a source of information for use by senior management. The OSC independently reviews a wide variety of design and configuration control-related activities, including proposed tests and experiments that affect nuclear safety, procedure changes, proposed changes to the Technical Specifications, proposed changes or modifications to plant systems or equipment that affect nuclear safety, investigation of violations of the Technical Specifications, 10 CFR 50.59 screenings and evaluations, and reportable events. OSC members have expertise in nuclear engineering, nuclear power plant testing, nuclear power plant operations, radiological safety, maintenance, technical advisory engineering, chemistry, quality control, and instrumentation and control with at least one (1) member qualified in each area. By conducting these reviews and reporting to the General Manager, Nuclear Operations or his designee, the OSC provides another check on the consistency of proposed activities with the BVPS design bases.

E.6.4.2 Independent Safety Evaluation Group

The Independent Safety Evaluation Group performs independent reviews of station activities, programs, procedures, modifications, and other activities in order to advise management on overall quality and safety of BVPS. Included in these reviews are root cause evaluations, special investigations and assessments to provide independent verification that activities are performed correctly and that human errors are reduced. ISEG independent root cause evaluations are conducted for reactor trips, unplanned loss of power to an emergency bus during power operation, common mode failures affecting

redundant safety-related components, safety injections, component failures causing a forced outage in excess of 10 days or other significant issues identified by management.

The Independent Safety Evaluation Group activities include reviews of design and configuration control issues. From 1992 to 1996 a total of thirty-seven (37) Independent Safety Evaluation Group reports dealt with design and configuration control issues. As issues were identified, assignments were made and corrective actions have been or are in the process of being implemented.

Because the Independent Safety Evaluation Group functions as an additional independent review activity which does not take the place of any other review organization that has responsibilities for nuclear safety-related activities, it provides yet another barrier to deviations from design bases requirements.

Moreover, because the Independent Safety Evaluation Group makes detailed recommendations to management, up to and including the Senior Vice President, Nuclear Power Division, on means to improve Unit safety and reliability, including equipment modifications, procedure revisions, maintenance activities and operations activities, the Independent Safety Evaluation Group influences management decisions. The recommendations initiated as a result of Independent Safety Evaluation Group evaluations are tracked to closure. Accordingly, the Independent Safety Evaluation Group provides additional support for the overall effectiveness of the processes for maintaining the plant consistent with its design bases.

E.6.4.3 Nuclear Safety Review Board

In 1996, at the direction of the Senior Vice President, Nuclear Power Division and in response to a self-identified need for a higher level of management involvement in the oversight and review of key issues, an NSRB was established. The creation of the NSRB was to help ensure that issues are thoroughly assessed, and appropriate actions taken, in the conduct of operations at BVPS. The NSRB independently reviews a number of design and configuration control-related activities including:

- Unreviewed Safety Questions
- Justification for Continued Operation (Enforcement Discretion requests)
- License amendments, including Technical Specification amendment requests
- Licensee Event Reports
- Responses to NRC Notice of Violations
- Category 1 and Category 2 Condition Reports

This additional layer of management oversight is expected to further strengthen the overall effectiveness of the process for maintaining the plant consistent with its design bases.

E.6.4.4 Offsite Review Committee

The ORC was established to independently review and audit matters that involve safety considerations relating to the operation of BVPS. The primary purpose of the ORC is to ensure that the BVPS Units are operated in a manner consistent with the terms of their operating licenses and in accordance with applicable regulations that are designed to safeguard the health and well-being of site personnel and the general public.

The ORC consists of a minimum of five (5) individuals who are appointed by the Senior Vice President, Nuclear Power Division to provide independent review and advice on safety considerations relating to operation of the Units. Consultants are also used to provide expert advice to the ORC. The ORC is required, by the operating license, to review matters involving safety considerations which include:

- Safety evaluations for changes to procedures, equipment or systems,
- Tests or experiments completed under the provision of 10 CFR 50.59,
- Proposed changes in Technical Specifications or licenses,
- Reportable occurrences requiring notification to the NRC,
- Recognized indications of an unanticipated deficiency in some aspect of design or operation of safety-related SSCs, and,
- Reports and meeting minutes of the OSC.

The ORC is also responsible for auditing to verify that BVPS is operated within licensed requirements. As discussed earlier in this Section, the consultant ORC members participate as Quality Services Unit audit team members. By participating in these audits, the ORC maintains current knowledge of the effectiveness of implementation of DLC design and configuration control programs.

This independent review by experienced senior management and consultant personnel provides additional assurance that programs and procedures established at BVPS maintain the facility consistent with its design bases.

E.7 EXTERNAL OVERSIGHT

E.7.1 Cooperative Management Assessment Program

As a charter member of the Nuclear Industry's Cooperative Management Assessment Program (CMAP) for the past 20 years, DLC has exchanged services of auditing personnel with other member utilities. Audits conducted by this organization assess the effective implementation of a utility's Quality Assurance Program. These audits serve as a management tool for determining the status and adequacy of a utility's quality activities in relation to meeting 10 CFR 50, Appendix B, including those related to plant design bases and configuration control.

E.7.2 Review of NRC Inspection Reports

The special and routine NRC inspections conducted over time have not only served to help ensure that BVPS is operated in accordance with the operating licenses and established regulatory requirements, but has also provided DLC with additional assurance that current processes and programs are consistent with design bases. Although these inspections have identified various deficiencies, implementation of effective corrective actions to improve and enhance the processes contribute to additional confidence in the overall effectiveness of these programs. A review of BVPS NRC SALP Reports from 1981 through 1996 and NRC Inspection Reports from 1987 through 1996 was performed by an external, independent review team to determine if significant trends exist involving the BVPS design bases programs. This review concluded that no noticeable trends were evident. This review also concluded that, in general, corrective actions were considered to be effective due to the prevention of recurring deficiencies.

E.8 RECENT ACTIONS RELATED TO DESIGN AND CONFIGURATION CONTROL

Three (3) recent events have been evaluated for their potential contributions to the analysis and conclusions in this Response. One event concerned the operation of two (2) of the three (3) Pressurizer Power Operated Relief Valves (PORV) Block Valves at BVPS 1 in the closed position since 1981. The second event involved inadequate control of a vendor who repaired a leaking valve in the Reactor Head Vent System.[6] The third event involved missing flood seals for the BVPS 2 Recirculation Spray pumps. A brief discussion of each event and its relation to this Response is provided below.

The long-standing misalignment of the PORV Block Valves is related to the NRC's request for information about the configuration and operation of SSCs consistent with their design bases. An investigation into this event determined that the event was attributable to the now superseded design and work control processes which did not adequately support return of the valves to their proper position after downstream equipment had been modified to meet seismic requirements.

The relation of the Reactor Head Vent valve repair event to the analysis is more complicated because the root cause stemmed from contractor control issues. Nevertheless, DLC does recognize that there was a failure to document that the Reactor Coolant System pressure boundary had been preserved in accordance with the Unit's design. Although the pressure boundary was found to have been preserved in accordance with design, the contributors to this event are of concern to DLC and corrective actions have been initiated to prevent their recurrence.

The missing flood seals for the Recirculation Spray pumps at BVPS 2 represents an inconsistency between the UFSAR and the physical plant and, as such, is also related to this Response. A root cause analysis determined that the missing seals likely resulted

[6] The closed Pressurizer PORV's block valves and the leaking Reactor Head Vent System Valve resulted in potential NRC escalated enforcement actions.

from confusion surrounding two (2) original construction documents. Corrective actions have been or are in the process of being implemented and include the development of a periodic inspection program for flood seals at BVPS 1 and BVPS 2.

DLC understands the significance of each of these events as demonstrated by its prompt and comprehensive corrective actions. DLC finds that the broader implications of the events are comparable to those associated with other situations that have been factored into this analysis. Further, these events do not, either by themselves or cumulatively with the other issues considered, such as those identified by the vertical slice reviews, change the analysis that supports a finding of reasonable assurance.

E.9 CONCLUSION

The programs and processes in place at BVPS, combined with DLC management's clear communication of expectations that the BVPS configuration be consistent with the design bases, implementation of a continuing critical self-checking process, the results of assessments conducted to date and analyses of recent plant events provide reasonable assurance that the configuration of BVPS 1 and BVPS 2 is consistent with the design bases.

F. RESPONSE TO REQUEST FOR INFORMATION REGARDING DESIGN BASIS RECONSTITUTION PROGRAMS AND RELATED ACTIVITIES

In responding to items a) through e), 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.

F.1 PAST DESIGN REVIEW AND RECONSTITUTION PROGRAMS

BVPS has undertaken many design review and/or reconstitution programs using a wide variety of formats. These design reviews and reconstitution programs are discussed in detail throughout Sections A through E of this Response and include the following:

- Vertical Slice Reviews (e.g., SSFEs, SWOPI, SSFI)
- Focused Vertical Slice Reviews
- Design Basis Document Program
- Design Basis Reconstitution Procedure
- Focused Design Reviews
- Drawing Reconciliation Program
- High-Energy Line Break Program
- Electrical Calculation Upgrade Program
- Seismic Qualification
- Environmental Qualification
- Fire Protection (Appendix R)
- Containment Penetrations

Reference is made to Sections A through E of this Response for a detailed description of these programs, including applicable structures, systems and components, plant-level design attributes, etc.

F.2 FUTURE DESIGN REVIEW AND RECONSTITUTION ACTIVITIES

DLC has initiated, or intends to initiate, the following activities designed to provide further assurance that BVPS is being operated and maintained within its design bases. To ensure a clear understanding of the commitments DLC is making in connection with this Response, they are delineated as follows:

1. Vertical Slice-type assessments
2. UFSAR Verification
3. Effectiveness Review of the Condition Report Program

F.2.1 Vertical Slice Review Type Assessments

DLC intends to conduct at least one (1) vertical slice-type assessment at each Unit every other year, starting in 1997 at BVPS 2 and in 1998 at BVPS 1.

F.2.2 UFSAR Verification

DLC has committed to conduct a detailed review of the UFSARs for both BVPS 1 and BVPS 2. The goal of this project will be to ensure compliance with the UFSAR. As previously docketed by letter dated December 26, 1996, this review is scheduled to be completed by December 31, 1998.

F.2.3 Effectiveness Review of Condition Report Program

As discussed in Section D, DLC will conduct a follow-up effectiveness review of the new Condition Report Program in 1997.

G. APPENDIX A

G.1 DEFINITIONS

Definitions of key terms that are used in the Response and/or certain other definitions are provided below. These are provided for clarification of meanings that may be unique to BVPS.

ADMINISTRATIVE CHANGE. A change to procedures which govern plant operations including system operations, maintenance, testing, off-normal conditions, emergency events, calibration, radiation control, and chemical-radiochemical control.

ADMINISTRATIVE CONTROLS. Rules, orders, instructions, procedures, policies, practices and designations of authority and responsibilities required by the DLC Operations Quality Assurance (QA) Program to obtain assurance of safety and high-quality operation and maintenance of BVPS.

ADMINISTRATIVE PROCEDURES. Those procedures used to establish administrative requirements for operation of BVPS as necessary to implement upper tier documents (e.g., 10 CFR, UFSAR, Technical Specifications, Operating License, BVPS QA Manual, Nuclear Power Division Administrative Manual). Administrative procedures are issued at the Nuclear Power Division, Unit, Department and Section levels.

BASELINE. (1) A specification or product that has been formally reviewed and agreed upon, that thereafter serves as the basis for further development, and that can be changed only through formal change control procedures. (2) A document or a set of such documents formally designated and fixed at a specific time during the life cycle of a configuration item. NOTE: Baselines, plus approved changes from those baselines, constitute the current configuration identification. (3) Any agreement or the result designated and fixed at a given time, from which changes require justification and approval.

BASELINE DOCUMENTS. The set of controlled documents that completely defines, describes, or implements the functional and physical characteristics of plant structures, systems, components and software which have been selected to be included in the document portion of the configuration management process.

CLEARANCE. The process used to denote that equipment has been placed in a safe condition by authorized DLC personnel to permit work, inspections, tests, etc..

COMPONENT. An assembly of interconnected parts that constitute an identifiable device, instrument, or piece of equipment. A component can be disconnected, removed as a unit, and replaced with a spare. It has definable performance characteristics that permit it to be tested as a unit. A component can be a card, a drawout circuit breaker, or other subassembly of a larger device, provided it meets the requirements of this

definition. As defined in ASME Section XI, a component includes: vessels, containments, piping systems, pumps, valves, core support structures, and storage tanks including their respective supports.

CONDITION REPORT. A report which provides a formal method to document any condition which does not meet expectations regarding materials, parts, components, activities, processes, procedures, and documents associated with the design, maintenance, or operation of either unit.

CONFIGURATION BASELINE. The set of controlled configuration items and baseline documents which define, describe, or implement the physical arrangement and functional characteristics of the plant.

CONFIGURATION CONTROL. (1) The process of evaluating, approving or disapproving, and coordinating changes to configuration items after formal establishment of their configuring identification. (2) The systematic evaluation, coordination, approval or disapproval, and implementation of approved changes in the configuration of a configuration item after formal establishment of its configuration identification.

CONFIGURATION ITEMS. Structures, systems, components, simulators, computer hardware and software which have been selected to be included in the equipment portion of the configuration management scope and are defined in the baseline documents.

CONFIGURATION MANAGEMENT. An integrated management process used to identify, control, provide status and verify the configuration baseline of the plant.

CONTROLLED DISTRIBUTION. Process of forwarding copies of documents to a specified set of recipients who are required to file the latest revision of the received documents in a prescribed arrangement, to destroy or remove any documents voided or superseded by the received documents, and to sign, date, and return to the distributing agency an acknowledgment of receipt.

CONTROLLED DOCUMENT. A controlled document (may be an individual procedure, set of procedures, or a manual) that describes and/or controls an activity or process and is issued to specific individuals or groups who are required to operate the plant, manage Design Change Packages, engineer drawings, and basically perform functions that would effect the operation of the plant. These individuals or groups need the most up-to-date and current information that is available. If revised, controlled documents are reissued to the same individuals or groups. Controlled documents are stamped as such.

CORRECTIVE ACTION. Action taken to determine the cause of one or more nonconforming condition(s) and where applicable implement measures to correct the condition and thereby preclude or minimize recurrence of the nonconforming condition.

CORRECTIVE MAINTENANCE. Maintenance action taken through the maintenance work request process to correct any plant equipment failure, deficiency or degraded

performance and restore equipment or systems to perform their originally designed functions.

DEFICIENCY. A condition that does not meet specified standards and requires corrective action.

DESIGN BASES DOCUMENTS (DBDs). Controlled documents used for referencing system and topical-related design bases information such as design specifications, drawings, calculations, licensing requirements, and modifications.

DESIGN BASES. That information which identifies the specific functions to be performed by a structure, system or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be restraints derived from generally accepted "state of the art" practices for achieving functional goals, or requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. The design bases of a facility, as so defined, is a subset of the licensing basis and is contained in the UFSAR.

DESIGN CHANGE. A modification to plant structures, systems, or components that changes design requirements that govern performance of the items' design bases.

DESIGN CHANGE PACKAGE (DCP). The collected documentation used to control the design change process in accordance with QA Program Procedure OP-4. Since the DCP is developed as the design change progresses, its contents at any given time are dictated by the degree to which the design change is complete.

DESIGN DOCUMENT. A document belonging to the set of documents comprised of design input documents, design studies or analyses, and design output documents that specify the design of a structure, system, or component. These are the documents to which one can refer to verify that structures, systems, and components have been designed to perform their intended function within the reference bounds of the controlling parameters and that form the point of reference for future plant modifications. Collectively, design documents encompass design bases information.

DESIGN EQUIVALENT CHANGE. A modification to plant structures, systems or components and/or design documents that does not change design requirements that govern performance of the items' design bases.

DESIGN INPUT. Those criteria, parameters, bases, or other design requirements upon which the detailed final design is based (ANSI N45.2.11-1974).

DESIGN OUTPUT. Documents such as drawings, specifications, and other documents defining the technical requirements of structures, systems, and components (ANSI N45.2.11-1974).

DESIGN PROCESS. Documented design practices such as calculations, analyses, evaluations, technical review checklists, or other documented engineering activities that substantiate the final design.

DEVIATION. A nonconformance or departure of a characteristic from specified requirements.

DOCUMENT. Any written, graphic, or pictorial information describing, defining, specifying, reporting, or certifying activities, requirements, procedures, or results.

EMPLOYEE CONCERN RESOLUTION PROGRAM (ECRP). The ECRP supplants and expands the Quality Concern Resolution Program and is directed at the resolution of nuclear safety concerns, quality concerns or other matters that may result in the violation of laws or regulations. This program is intended to provide a confidential communications process to avoid fear of potential retaliation.

ENGINEERING DOCUMENT. A document, such as specifications, calculations, sketches, design concepts, design drawings, vendor information, etc., which is approved by the Nuclear Engineering Department and identified by a DLC Document Number.

ENGINEERING MEMORANDUM (EM). A document that provides engineering information or identifies and provides a response to a question concerning a design change, station operation, or a component or system problem for which a modification may be required or desired.

EQUIPMENT IDENTIFICATION NUMBER (EIN). A combination of numbers, letters, and/or special characters which identify plant equipment and components. EINs are commonly known as mark numbers, line numbers, support numbers, or penetration numbers depending on the application.

EQUIPMENT QUALIFICATION (EQ). The generation and maintenance of documentary evidence to demonstrate and assure that equipment within the scope of 10 CFR 50.49 is capable of operating on demand to meet system performance requirements, including subjection to adverse environmental conditions.

EQUIPMENT TECHNICAL INFORMATION. Includes as a minimum:

- Vendor-supplied engineering and technical information (e.g., drawings, manuals, technical bulletins) and changes thereto,
- Equipment Qualification data (provided by the equipment vendor or qualification lab), and
- Industry-developed information, including utility and NRC-originated information (NPRDS, SER, IEB, IEN, etc.).

INDEPENDENT REVIEW. A review completed by personnel not having assigned responsibility for the preparation of the document under review regardless of whether

they operate as part of the Nuclear Power Division organization or as individual staff members.

INSTALLATION ORGANIZATION. The DLC organization responsible for performing the work described in a work document or Design Change Package. This normally is the Nuclear Construction Department or the Station Maintenance Group.

MAINTENANCE WORK REQUEST (MWR). A work package used to report deficiencies which require corrective or planned maintenance and document corrective action taken. QA Category I and Category F MWRs are permanent QA records.

MASTER EQUIPMENT LIST (MEL). A computer data base which contains information for plant equipment and components that have been assigned equipment identification numbers.

MODIFICATION. Any physical or functional change to a site facility or baseline document.

NONCONFORMING CONDITION. A deficiency in a characteristic, documentation or procedure which renders the quality of an item or process unacceptable or indeterminate (e.g., physical defects, test failures, and incorrect or inadequate documentation or deviation from prescribed processing, inspection or test procedures).

NON-INTENT PROCEDURE CHANGE. The only changes which do not alter procedure intent are:

1. Spelling, punctuation, and editorial.
2. Updating of procedural references which do not alter content of the procedure.
3. Conforming to previous OSC approved changes.
4. Addition to or substitution of equivalent items and test equipment that will not change the required minimum accuracy of the measurement or method of testing.
5. Addition of industrial safety instructions.
6. Addition of approved administrative instructions (procedure specific).
7. Revise or rearrange instruction steps to clarify procedure performance, activity or influence as defined by the Procedure, Purpose, Objective or Acceptance Criteria provided the change does not deviate from equipment technical information, technical specifications, CFR, UFSAR, QA Program, Station Commitment, and other regulatory requirements.

8. Addition of steps(s) which were an obvious omission for completing a procedure loop or cycle (e.g., reinstalling a removed component or removing an installed component that is identified to be reinstalled).

NUCLEAR SAFETY REVIEW BOARD (NSRB). A high-level management committee appointed by the Senior Vice President, Nuclear Power Division to provide oversight on matters related to nuclear safety.

OFFSITE REVIEW COMMITTEE (ORC). A committee appointed by the Senior Vice President, Nuclear Power Division to provide independent review and audit of designated activities.

ONSITE SAFETY COMMITTEE (OSC). A committee established to advise the Operational Managers on matters related to nuclear safety, with review capability in designated areas.

OPERATIONAL ACCEPTANCE. The partial or complete acceptance of a Design Change Package by the General Manager, Nuclear Operations or his designee for service following the successful completion of proof, functional, and operational tests and the required turnover checklist items.

OPERATOR AID. Any print, table, graph, handwriting or other posting or pen marking that may be used by any site personnel involved in facility operations to make a safety related decision while performing their job function.

PERFORMANCE TEST. A test of a system or component to verify that required performance characteristics are being achieved.

PLANT INSTALLATION PROCESS STANDARDS (PIPS). A series of standards which combine generic technical requirements with installation procedural and inspection requirements to control and standardize specific BVPS installation activities performed either by the Construction organization or the Station.

POST MAINTENANCE TESTING. Tests performed after maintenance to demonstrate that the repair was successful in restoring the equipment to full operability.

PROBLEM REPORT. A report which provides a formal method to document an event or incident that has been determined by an individual to have an impact on the operation of BVPS. These reports are used to determine the cause of the problem and initiate corrective actions.

QA CATEGORY 1. Plant systems, or portions of systems, structures and equipment whose failure or malfunction could cause a release of radioactivity that would endanger public safety. This category also includes equipment which is vital to a safe shutdown of the station and the removal of decay heat and sensible heat, or equipment which is necessary to prevent or mitigate consequences to the public of a postulated accident.

QA CATEGORY F. The Quality Assurance category assigned to structures, systems, barriers, and components designed to detect, suppress, and prevent the spreading of a fire that can affect such plant areas and equipment necessary to assure:

- the acceptable performance of necessary safe shutdown functions,
- the protection of power generation systems and equipment, and
- the maximum protection against radioactive release and the least risk to public health and safety.

RECORD. Any information, regardless of media, maintained to provide documentary evidence of the characteristics of an activity, structure, system or component, and of the operations affecting those characteristics and their development.

RETRIEVABLE DOCUMENTS. Documents not readily accessible but that can be located in utility or contractor files or archives and that contain information that is valid for use in the design or design change process.

SAFETY EVALUATION. A written evaluation performed to make a change to the facility or the procedures described in the UFSAR, or to conduct tests or experiments not described in the UFSAR. The safety evaluation is required by 10 CFR 50.59 and is performed to determine if an unreviewed safety question is involved.

SAFETY RELATED. Those plant features (e.g., structures, systems, components) necessary to assure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents which could result in offsite exposures comparable to the guideline exposures of 10 CFR 100 (see also Category I).

SETPOINT. The point(s) at which a device changes state to cause a function; this includes the tolerance to the nominal setting.

STRUCTURE. A nuclear power reactor structure consists of parts that are put together to form safety-related barriers, containments (e.g., buildings and housings), and enclosures that provide a partition between reactor systems and components and the environment to prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

SYSTEM. An integral portion of a nuclear unit composing electrical, electronic, or mechanical components (or combinations thereof) that may be operated as a separate entity to perform a particular function. Specifically, a set of components, parts or pieces that form an entity for a general protective function.

TECHNICAL EVALUATION REPORT (TER). A report used to document and track any engineering task of an evaluative nature that is not controlled by other existing processes.

TEMPORARY MODIFICATION. Temporary minor alterations made to plant equipment, components or systems that do not conform with approved drawings or other design documents. These alterations are temporary in that they are expected to be installed for a limited period of time based on management review and approval and include leak repairs temporary jumpers, lifted leads, bypasses, blocks, and temporary gauges.

TRAIN. A collection of equipment that is configured and operated to serve some specific Unit safety function and may be a subset of a system.

VENDOR TECHNICAL INFORMATION. The documentation provided by a vendor that provides technical information regarding the design, installation, operation, maintenance, testing or qualification of structures, systems, or components such as vendor drawings and vendor technical manuals.

H. APPENDIX B

H.1 ACRONYMS

AFWS	Auxiliary Feedwater System
ANSI	American Nuclear Standards Institute
ASME	American Society of Mechanical Engineers
AMSAC	Anticipated Transients Without Scram Mitigating System Activation Circuitry
BVPS	Beaver Valley Power Station, Units No. 1 and No. 2
BVPS 1	Beaver Valley Power Station, Unit No. 1
BVPS 2	Beaver Valley Power Station, Unit No. 2
CFR	Code of Federal Regulations
DBD	Design Basis Document
DCP	Design Change Package
DLC	Duquesne Light Company
EOP	Emergency Operating Procedure
EPRI	Electrical Power Research Institute
FSAR	Final Safety Analysis Report
INPO	Institute of Nuclear Power Operations
MOV	Motor Operated Valve
NRC	United States Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSRB	Nuclear Safety Review Board
NUMARC	Nuclear Management and Resource Council
ORC	Offsite Review Committee
OSC	Onsite Safety Committee
QSU	Quality Services Unit
RVLIS	Reactor Vessel Level Instrumentation System
SER	Safety Evaluation Report
SSC	Structure, System and Component
SSFE	Safety System Functional Evaluation
SSFI	Safety System Functional Inspection

SWOPI	Service Water Operational Performance Inspection
UFSAR	Updated Final Safety Analysis Report