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Beaver Valley Power Station Unit 1
Operating License No. DPR-66
Docket No. 50-334

Beaver Valley Power Station Unit 2
Operating License No. NPF-73
Docket No. 50-412

Davis-Besse Nuclear Power Station
Operating License No. NPF-3
Docket No. 50-346

Perry Nuclear Power Plant
Operating License No. NPF-58
Docket No. 50-440

Subject: Response to Request for Additional Information (RAI) Regarding Proposed FirstEnergy Nuclear Operating Company Quality Assurance Program Manual Revision 1 (TAC Nos. MB0914 and MB0915)

Ladies and Gentlemen:

By letter dated December 27, 2000, the FirstEnergy Nuclear Operating Company (FENOC) requested Nuclear Regulatory Commission (NRC) review and approval of Revision 1 of the FENOC Quality Assurance Program Manual. By letter dated April 12, 2001, the NRC requested additional information necessary to complete its review of this revision. The following attachments are provided in response to that request:

Attachment 1 provides responses to Items 1 through 10 of the subject RAI.

Attachment 2 is a marked-up version of the FENOC QAPM reflecting changes made as a result of the RAI. To simplify NRC review, paragraph A.1.c of the attachment has also been revised to incorporate a commitment to apply the QAPM to activities related to FENOC power plants subject to 10 CFR 72 (currently, only the Davis-Besse Nuclear Power Station).

Attachments 3 through 6 contain Safety Analysis Report (SAR) markups and existing SAR descriptions that identify how regulatory guide commitments would be incorporated in the respective SARs for each of the FENOC power plants. These attachments are associated with the response to Item 6 of the RAI.

Q004

If you have questions or require additional information, please contact William R. Kanda, Director of Oversight and Process Improvement at (440) 280-5579.

Very truly yours,

Robert J. Saunders

Attachments
Enclosure

cc: NRC Project Manager, Beaver Valley Power Station
NRC Project Manager, Davis-Besse Nuclear Power Station
NRC Project Manager, Perry Nuclear Power Plant
NRC Senior Resident Inspector, Beaver Valley Power Station
NRC Senior Resident Inspector, Davis-Besse Nuclear Power Station
NRC Senior Resident Inspector, Perry Nuclear Power Plant
NRC Region I
NRC Region III
Utility Radiological Safety Board

ATTACHMENT 1

Responses to Items 1 through 10 of NRC RAI Dated April 12, 2001

NRC RAI Question 1:

The Direct Final Rule, effective on April 26, 1999 (64 FR 9030), revises the regulation of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.54(a)(3) to provide additional flexibility for licensees making changes to their QAP without obtaining Nuclear Regulatory Commission (NRC) approval of these changes in advance. Forty-nine changes to the QAP have been identified in Attachment 4 of the submittal. Please identify those changes considered to be reductions in commitments.

Response:

BVPS has identified the following items as reductions in commitments (RICs): 14, 24, 30, 32, 44 and 46.

Perry has identified the following items as RICs: 11, 12, 14, 15, 17, 19, 20, 21, 23, 24, 27, 28, 29, 30, 32, 33, 34, 37, 39, 42, 43, 44, 45, 46, 47, and 49.

Davis-Besse has identified the following as RICs: 11, 12, 14, 15, 17, 19, 20, 21, 23, 24, 27, 28, 29, 30, 32, 33, 34, 37, 39, 42, 43, 44, 45, 46, 48 and 49.

NRC RAI Question 2:

The QAPM revision creates a corporate QA oversight function (Attachment 3, Section A.2.a). Please clarify whether this function implements the guidance of ANSI N18.7 – 1976, Section 4.3, for independent review. Clarify how the provisions of Section 4.3 are implemented. Describe any reporting relationships between the sites and the corporate function. (Side Bar 3)

Response:

Revision 0 of the FENOC Quality Assurance Program Manual, as currently implemented by Davis-Besse and Perry, established the corporate function for establishing the policies, goals, and objectives for the FirstEnergy Nuclear Operating Company (FENOC), and implementing and controlling the FENOC QAPM (paragraph A.2.a.1). The Director of this organization also functions as the independent review committee chairman, which includes responsibility for the administration of the committee's activities. This function reports to the chief executive officer (President and Chief Nuclear Officer of FENOC) as described in paragraph A.2.a. During the development of Revision 1 of the FENOC QAPM this function was evolving, and it was decided to include the quality assurance oversight function.

Revision 1 therefore moved the discussion for quality assurance oversight from its current reporting relationship at the site level (paragraph A.2.c.1) under paragraph A.2.a.1 (as described above). With this move, it was intended that the quality assurance oversight function (i.e., audits) would be a corporate function with dedicated QA organizations physically located at each site. In addition to the site level traditional QA

oversight function, this organizational alignment would facilitate a collective assessment function to improve performance at all sites consistently.

This organization and the associated functions fulfill the requirements of ANSI N18.7-1976 for the independent review and audit programs. As the QA oversight function, the organization satisfies the requirements of Section 4.4. As the independent review committee function, the Director satisfies the requirements of Section 4.3. This includes ownership of the committee's charter (Section 4.2.1) and administration of the committee's activities (Sections 4.2.2, 4.2.3, 4.2.4, and 4.2.5). Individuals assigned to the committee satisfy the qualification requirements of Section 4.2 and ANSI N18.1. Subjects reviewed by the committee satisfy the requirements of Section 4.3.

The Director of this organization reports to the President and Chief Nuclear Officer. The QA Managers located at the sites report to the Director, but have access to site management for purposes of conducting audits and assessments. Individuals associated with paragraphs A.2.a.1 and A.2.a.1.a) are independent from cost and schedule when opposed to safety considerations. The independent review committee is comprised of both on-site and off-site individuals, and serves all sites.

Paragraph A.2.a.1 has been modified to specifically identify the individual responsible for independent reviews (Refer to revised Attachment 2. The last sentence in the paragraph has been added.).

NRC RAI Question 3:

Changes in the commitment to Regulatory Guide (RG) 1.8, "Personnel Selection and Training," should continue to be subject to the review requirements of 10 CFR 50.54(a). RG 1.8 is explicitly referenced by Standard Review Plan 17.3 as applicable to the QA regulations of Appendix B. (Side Bar 6)

Response:

The wording deletion was not intended to reflect any change in the application of RG 1.8, "Personnel Selection and Training." The regulatory guide or its equivalent is contained as a requirement in each plant's technical specifications. Therefore, a description of RG 1.8 compliance has been included in the QAPM. (Refer to revised Attachment 2, Table 1.)

NRC RAI Question 4:

The original wording "the guidance will be applied to activities comparable in nature and extent to construction phase activities," is clearer than the proposed clarification (Attachment 3, Section A.7.a.5) and should be retained. (Side Bar 9)

Response:

The wording change was intended to provide clarity and was not intended to reflect any change in the application of regulatory guidance to plant operations. To ensure clarity, the wording in Section A.7.a.5 has been revised (Refer to revised Attachment 2).

NRC RAI Question 5:

The revised QAPM differentiates between audits that satisfy Appendix B requirements and those that do not. Identify which of the audits in Section C.2.a satisfies Appendix B requirements. Clarify any differences in QAP implementation for the two audit types, particularly with respect to RG 1.144 and 1.146. (Side Bar 16)

Response:

Side Bar 16 indicates that not all of the subsequently listed audits are Appendix B audits. This statement was added because the list contained only those audits that had been located in technical specifications. Technical specifications identified these as audits performed under the cognizance of the independent review committee. In practice, many of these audits were performed by the QA organization at the request of the independent review committee because these auditing resources were readily available. However, the audits were not necessarily being performed for the purpose of satisfying Appendix B, particularly when they examined activities that were not subject to Appendix B. Since the QAPM describes an Appendix B QA program, it was expected that readers would naturally infer that the listed audits are all related to Appendix B unless a clarifying statement was provided. The statement was also added to prevent possible misconception that all of the audited topic areas and activities are subject to Appendix B controls.

RGs 1.144 and 1.146 describe acceptable methods for auditing of Appendix B QA programs. These must be applied to audits of the Appendix B program and when a commitment has been made to apply the Appendix B QA program to satisfy non-Appendix B QA requirements, such as for dry fuel storage. Although not required by the license, FENOC commitments to these guides would usually be extended to other audits (i.e. non-Appendix B audits) because there is normally no practical advantage to deviating from prevailing practices of the auditing organization. Therefore, paragraph C.2.a.2 (Refer to revised Attachment 2) has been revised to remove the statement in question.

NRC RAI Question 6:

Commitments to the following regulatory guides (and associated standards) are proposed to be removed from the QAPM: RG 1.26, RG 1.29, RG 1.54, RG 1.55, RG 4.15, and RG 1.78. If these commitments are duplicated elsewhere in the Final Safety Analysis Reports (FSARs), please provide the reference to applicable FSAR sections for each plant. If these commitments are to be relocated in conjunction with the QAPM

revision, identify the FSAR sections where they will be relocated, provide marked-up pages indicating any revisions that will be made to the relocated QAPM revision. Cite instances where an NRC safety evaluation has approved removal of any of these commitments to RGs from a licensee's QAP. (Side Bar 19, 21, 33, 34, 48, 49)

Response:

For those regulatory guides that are being relocated to the FSAR or for those that are already maintained in the FSAR, FSAR excerpts and/or markups are provided in Attachments 3 through 6. Text being relocated to FSARs will remain consistent with current positions applicable to each plant. FSAR change requests will be approved no later than the date for adoption of the revised QAPM at each plant. The version of the subject regulatory guides committed to at the various plants may vary, due to the different licensing dates.

Section 4.9 of the SER for ENTERGY (Letter to Mr. Michael R. Kansler dated November 6, 1998 [TAC No. M97893]) recognized increased emphasis on the licensee's commitments to regulatory guides due to removing duplicate material from the body of the QAPM. As a result, NRC staff reviewed the proposed regulatory guide positions in the QAPM against existing commitments to regulatory guides and associated standards for each of the ENTERGY plants. For several of the Entergy plants, some regulatory guide positions previously existing in plant-specific QA program descriptions were not subsequently invoked by the new QAPM. Thus the positions had been removed from the respective plant's QA program description. The NRC noted changes to regulatory guide positions that were considered significant reductions in commitments and the regulatory guide positions being deleted were not included in their discussion.

Beaver Valley-2

In Table 17.1, the original SER for BVPS-2 (NUREG 1057) the NRC listed the regulatory guidance applicable to the quality assurance program. Regulatory Guides 1.26, 1.29, 1.54, 1.55, 4.15 and 1.78 were not listed in this Table. Therefore, although BVPS-2 committed to all but RG 4.15, these commitments were not considered to be part of the quality assurance program description for Unit 2.

BVPS-2 UFSAR Table 1.8-1 contains the positions for Regulatory Guides 1.26, 1.29, 1.54, 1.55, and 1.78. Attachment 3 contains excerpted pages that describe existing positions.

Beaver Valley-1

BVPS-1 UFSAR Section 1.3 contains positions for Safety Guide 26, Safety Guide 29, and RG 1.54. RG 1.78 commitments are described in Sections 2.1.5 and 9.13.4. BVPS-1 has not committed to RG 1.55 or RG 4.15. Attachment 4 contains excerpted pages and markups related to this response.

Davis-Besse

Existing USAR Sections 3.2.1 and 3.2.2 already discuss seismic classification and system quality group classification. The proposed QAPM revision would not change these commitments, but references to related RGs 1.29 and 1.26 will be added. Aspects of RG 1.78 are discussed in USAR Sections 2.2.3.6, 6.4.2 and 15.4.8. The QAPM position will be relocated into USAR Section 6.4.2. Positions regarding RGs 1.54 and 4.15, not currently discussed in the USAR, will be added. Consistent with the existing QAPM position, no commitment has been made to RG 1.55. Attachment 5 contains excerpted pages and markups related to this response.

Perry

Table 1.8-1 of the Perry USAR will be revised to include relocated positions for RGs 1.26, 1.29, 1.54, 1.55 and 1.78. RG 4.15 is not listed in this Table and is not considered to be part of the quality assurance program description for PNPP. Attachment 6 contains excerpted pages and markups related to this response.

NRC RAI Question 7:

With respect to RG 1.30, the QAPM commitment is revised to meet the intent instead of the positions of the RG. Clarify the term "intent" by specifying the proposed exceptions or alternatives to RG 1.30. (Side Bar 22)

Response:

Although BVPS 2 was licensed to the position in the proposed Revision 1 of the FENOC QAPM, it was decided to maintain the existing FENOC position and apply it to all plants. Revised Attachment 2, Table 1 now includes the position.

NRC RAI Question 8:

The QAPM revision proposes an exception to the guidance of ANSI N18.7, Section 4.3 in that the license amendments will not be independently reviewed. The basis provided (Attachment 4) is that amendments are reviewed by an onsite review body (ANSI N18.7, Section 4.4). Since the onsite review body already reviews license amendment changes as part of its responsibilities, the exception effectively eliminates independent review. Provide additional justification for this exception. (Side Bar 24)

Response:

The proposed changes to the FENOC Quality Assurance Plan include the elimination of the requirement for the Company Nuclear Review Board (CNRB) to independently review all license amendment requests (LARs). The three FENOC plants each generate 10 to 20 LARs, on average, per year. Each LAR typically contains detailed system or analysis related information. The presentation and explanation of this information to the CNRB consumes a significant amount of the CNRB's time. In FENOC's experience, the detailed information contained within the LAR documentation now receives a

comparable independent review by plant organizations, as required by the plant administrative procedures governing LAR preparation, review, and approval prior to submittal to the NRC. Recognition of these independent reviews and the elimination of the CNRB independent review of LARs would allow the CNRB to focus on other more pertinent contemporary issues that warrant oversight.

The requirement for the CNRB to review all changes proposed to the Technical Specifications and Operating License is based on the standards contained in ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants" (the standard) which is endorsed by Regulatory Guide 1.33, " Quality Assurance Program Requirements." The specific review requirement is contained in Section 4.3.4 of the standard, " Subjects Requiring Independent Review." Section 4.3.4 of the standard lists the items that require independent review. Item (3) of Section 4.3.4 specifies "Changes in the technical specifications or license amendments relating to nuclear safety...". The standard defines an independent review as follows:

> "Review completed by personnel not having direct responsibility for the work function under review regardless of whether they operate as part of an organizational unit or as individual staff members..."

Each FENOC plant performs an extensive independent onsite review of LARs. The preparation of LAR documentation is typically the direct responsibility of the Regulatory Affairs section at each FENOC site. As a minimum, each LAR typically receives an independent review by engineering personnel with specific system or analysis background pertinent to the subject of the LAR, an Operations independent review, and an independent committee review by the onsite safety review committee. Additional reviews of the LAR are performed by Regulatory Affairs personnel and upper management prior to submitting the LAR to the NRC. As part of the proposed elimination of the CNRB required review of LARs, FENOC would commit to a more formal requirement that each LAR receive an independent review by engineering personnel and Operations personnel in addition to the currently required independent review by the onsite safety review committee.

The series of onsite reviews used to process an LAR ensures that each LAR receives a multidisciplinary independent review by qualified plant personnel. As such, the onsite review process fulfills the requirement for an independent review as defined in the standard. In addition, once submitted to the NRC, the LAR undergoes additional detailed review by the NRC staff. It is significant to note that unlike many other items reviewed by the CNRB, LARs are ultimately subject to NRC review and approval.

In Section 4.1 of the standard, where general review and audit information is discussed, the standard explains that,

> "This standard does not specify an organizational structure for meeting the review and audit functions, but in lieu thereof delineates essential elements of

> satisfactory comprehensive programs for review and for audit in the manner best suited to the owner organization involved."

Section 4.1 recognizes that "The programs provided for reviews and for audits may take different forms." The standard goes on to describe that,

> "Historically a committee approach was used to provide both review and audit capability for early commercial nuclear power plants. This approach was employed to make the most efficient use of people with pertinent experience and qualifications. In the ensuing period, the availability of competent personnel has significantly increased as the nuclear power industry has expanded and the sources of trained manpower have responded to the resulting demand. This growing pool of talent in the aggregate, is sufficient to encourage alternative approaches to the review and audit committees commonly used in the past."

The standard further recognized that expanding nuclear power owner organizations "should regard the use of committees to meet the independent review functions as an interim approach for effective utilization of available technical expertise." Accordingly, even as early as 1972 the standard recognized that the necessity of a standing committee to effectively share experienced manpower may no longer be necessary. Over the years the FENOC plants have developed a large extensive body of onsite technical expertise that is now routinely utilized for the review functions addressed in the standard.

In the early 1970's, during which time the standard was being developed, onsite staffing levels at plants being built were projected to typically be less than a hundred in total number. Today, however, the typical onsite technical and operating staff levels alone are several hundred. This much larger onsite staff provides a significant increase in the technical expertise available to review LARs. Factoring this growth of expertise available for use at each FENOC plant to independently review LARs provides a viable alternative to the CNRB's review of LARs.

In summary, the proposed change to the FENOC QAPM eliminates the requirement for the CNRB to review LARs. However, for LARs, the FENOC plants have developed the "alternative" approach to a standing "offsite" review committee as suggested by the standard. The utilization of onsite expertise, including an independent review by the onsite review committee, effectively meets the requirements of the standard and has eliminated the need for an additional CNRB review of LARs. Due to the level of review now provided by experienced onsite personnel, which is comparable to the level of review provided by the CNRB, the elimination of the CNRB review of LARs does not reduce the effectiveness of the FENOC QAPM nor does it affect how the plan complies with 10 CFR 50, Appendix B.

NRC RAI Question 9:

The revised commitment to RG 1.33 reduces the scope of applicability of QA requirements to structures, systems, and components (SSCs) that are "safety related."

Currently, the QAPM is applicable to SSCs that are "important to safety." The regulations do not always differentiate between the terms "important to safety" and "safety-related." For example, the General Design Criteria apply to SSCs important to safety, although the term "safety-related" is generally implied. IEEE Standard 279 which addressees protection systems, refers to important to safety functions: this standard is incorporated by reference into the regulations (10 CFR 50.55a(h)). Further, SSCs that are "important to safety", but not necessarily "safety-related" generally fall within the scope of most QAPs to an extent consistent with their importance to safety (Appendix B, Criterion II). Examples of nonsafety-related SSCs that are important to safety include those associated with systems designed to prevent or mitigate anticipated transients without scram, station blackout, and fire protection. A more complete discussion should be provided, which defines the scope of the FENOC QAP in general and the specific examples cited above. (Side Bar 26)

Response:

The change in wording in Table 1, Section B.2.a (now C.2.a) of the QAPM, from "functions important to the safety of nuclear power plant structures, systems and components" to "affecting the safety-related functions of nuclear power plant structures, systems and components" was made to reflect the wording in Section 1, "Scope" of ANSI N18.7-1976 which FENOC is committing to, and which is the ANSI standard endorsed by the NRC in RG 1.33, Revision 2. The wording in Revision 0 of the FENOC QAPM reflected the words in Section 1 of ANSI/ANS 3.2-1982, which FENOC had previously committed to, and which the NRC has not endorsed. Both Revision 0 and Revision 1 of this section of the QAPM are in quotation marks, since they reflect the actual wording of the ANSI standards referenced. The proposed change was not made to reflect any change in the scope of applicability of RG 1.33 to structures, systems and components at the FENOC plants. This change does not alter existing commitments regarding application of the QA program to areas such as systems designed to prevent or mitigate anticipated transients without scram, station blackout, and fire protection.

NRC RAI Question 10:

Procedural controls are generally included in plant technical specifications under administrative controls (10 CFR 50.36(a)(5)). The NRC has allowed licensees to relocate certain administrative controls, including those for procedure adherence, to the QAP. For each plant covered by the QAPM, identify the licensing basis document that includes the process for controlling temporary changes to procedures. For each plant, identify the regulatory process used for controlling these changes. Cite references to NRC safety evaluations that have approved control of these changes through the 10 CFR 50.59 change control process. (Side Bar 30)

Response:

QAPM Table 1, item B.2.c (now C.2.c, refer to revised Attachment 2) has been changed to reflect existing plant-specific processes for controlling temporary procedure changes.

NRC letter from Mr. Daniel Collins, Project Manager to Mr. Lew Myers dated January 12, 2000 (Subject: Beaver Valley 1 and 2 - Approval of Proposed Revisions to the Quality Assurance Program Description [TAC Nos. MA4992 AND MA4993]) provided NRC approval of the alternative described in Paragraph C.2.c.2 of the revised position.

ATTACHMENT 2

REVISED FENOC QAPM

FENOC

FirstEnergy Nuclear Operating Company

QUALITY ASSURANCE PROGRAM MANUAL

Beaver Valley Power Station
Units 1 & 2
Docket Nos. 50-334 & 50-412
Operating License Nos. DPR-66 & NPF-73

Davis-Besse Nuclear Power Station
Docket No. 50-346
Operating License No. NPF-3

Perry Nuclear Power Plant
Docket No. 50-440
Operating License No. NPF-58

Revision 1

FENOC
FirstEnergy Nuclear Operating Company

QUALITY ASSURANCE PROGRAM MANUAL

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A. MANAGEMENT

1. Methodology

a. The Quality Assurance Program Manual (QAPM) provides a consolidated overview of the quality program controls which govern the operation and maintenance of FirstEnergy Nuclear Operating Company's (FENOC's) quality related items and activities. The QAPM describes the quality assurance organizational structure, functional responsibilities, levels of authority, and interfaces.

b. The requirements and commitments contained in the QAPM are mandatory and must be implemented, enforced, and adhered to by all individuals and organizations. Employees are encouraged to actively participate in the continued development of the QAPM as well as its implementation. Changes are promptly communicated when identified.

c. The QAPM applies to all activities associated with structures, systems, and components which are safety related or controlled by 10 CFR 72. The requirements of the QAPM are applied to these items and activities to an extent commensurate with their importance to safety. The applicability of the requirements of the QAPM to other items and activities is determined on a case-by-case basis. The QAPM implements 10 CFR 50 Appendix B. It also applies to activities related to facilities licensed under 10 CFR 72, Subpart G.

d. The QAPM is implemented through the use of approved procedures (i.e., policies, directives, procedures, or other documents) which provide written guidance for the control of quality related activities and provide for the development of documentation to provide objective evidence of compliance.

2. Organization

The organizational structure responsible for implementation of the QAPM is described below. The specific organization titles for the quality assurance functions described are identified in procedures. The authority to accomplish the quality assurance functions described is delegated to the incumbent's staff as necessary to fulfill the identified responsibility.

a. The president and chief nuclear officer is responsible for providing top level direction of all activities associated with the safe and reliable operation of FENOC's nuclear sites. The president and chief nuclear officer provides guidance with regards to company quality assurance policy.

1. The individual responsible for oversight reports to the president and chief nuclear officer and is responsible for establishing the policies, goals, and objectives and the implementation of the quality assurance program of FENOC's corporate activities and maintaining this QAPM in accordance with regulatory requirements. This individual also has overall responsibility for the quality assurance and independent safety review committee functions.

a) The individual responsible for quality assurance reports to the individual responsible for oversight and has overall authority and responsibility for verifying the implementation and adequacy of the quality assurance program as described in this QAPM. The individual responsible for quality assurance has the authority and responsibility to escalate matters directly to the president and chief nuclear officer when needed.

b. The executive responsible for overall plant nuclear safety, operations support, and engineering at each site reports to the president and chief nuclear officer. This executive is responsible for establishing and maintaining policies, goals, and objectives of this QAPM at the respective site and overseeing activities of the off-site safety review committee.

c. The individuals fulfilling the following management functions report to the executive identified in Paragraph 2.b above. These individuals may report through an additional layer of management but shall maintain sufficient authority and organizational freedom to implement the assigned responsibilities. These individuals may be responsible for a single unit/location or for multiple units/locations and may fulfill more than one function described below. Conversely, responsibilities may be fulfilled by more than one individual.

1. The individual responsible for plant operations assures the safe, reliable, and efficient operation of the plant within the constraints of applicable regulatory requirements and the operating license.

2. The individual responsible for plant modification provides direction, control, and overall supervision of the implementation of plant modifications and assigned maintenance.

3. The individual responsible for training provides direction, control, and overall supervision of all training of personnel required by regulations.

4. The individual responsible for records management provides direction, control, and overall supervision of the records management program and associated activities.

5. The individual responsible for document control provides direction, control, and overall supervision of the document control program and associated activities.

6. The individual responsible for the corrective action program provides direction, control, and overall supervision of the corrective action program and associated activities.

7. The individual responsible for engineering is responsible for the development and maintenance of engineering programs, policies, and procedures and for providing engineering services.

8. The individual responsible for materials, purchasing, and contracts is responsible for supplier evaluations, source verifications, procurement, services, receipt, storage, and issue of materials, parts, and components.

9. The individual responsible for quality control has the responsibility for establishing, controlling, and implementing the quality control inspection program. The individual responsible for quality control has the authority and responsibility to escalate matters as needed.

d. The on-site and off-site safety review committees independently review activities to provide additional assurance that the units are operated and maintained in accordance with the Operating License and applicable regulations which address nuclear safety.

3. Responsibility

a. FENOC has the responsibility for the scope and implementation of an effective quality assurance program.

b. FENOC may delegate all or part of the activities of planning, establishing, and implementing the quality assurance program to others, but retains the responsibility for the program's effectiveness.

c. The adequacy of the QAPM's implementation is continually assessed by the individual(s) responsible for quality assurance and the associated executive for overall plant nuclear safety, and is reported to the individual responsible for oversight and to the president and chief nuclear officer.

d. FENOC is responsible for ensuring that the applicable portion(s) of the quality assurance program is properly documented, approved, and implemented (people are trained and resources are available) before an activity within the scope of the QAPM is undertaken by FENOC or by others.

e. Responsible individuals are to ensure that personnel working under their management cognizance are provided the necessary training and resources to accomplish their assigned tasks within the scope of the QAPM.

f. Procedures that implement the QAPM are approved by the management responsible for the applicable quality function. These procedures are to reflect the QAPM and work is to be accomplished in accordance with them.

4. Authority

a. When FENOC delegates responsibility for planning, establishing, or implementing any part of the overall QA program, sufficient authority to accomplish the assigned responsibilities is delegated.

b. The individual responsible for quality assurance has the responsibility and the authority to stop unsatisfactory work (including reactor operation through proper channels) and control further processing, delivery, installation, or use of non-conforming items or services. Cost and schedule considerations will not override safety considerations.

5. Personnel Training and Qualification

a. Personnel assigned to implement elements of the quality assurance program are capable of performing their assigned tasks.

b. Training programs are established and implemented to ensure that personnel achieve and maintain suitable proficiency.

c. Personnel training and qualification records are maintained in accordance with procedures.

d. Additional details concerning personnel training and qualification may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.8, 1.58, and 1.146).

6. Corrective Action

a. It is the responsibility of each individual to promptly identify and report conditions adverse to quality. Management at all levels encourages the identification of conditions that are adverse to quality.

b. A corrective action program is established and implemented that includes prompt identification, documentation, significance evaluation, and correction of conditions adverse to quality. For significant conditions adverse to quality, the cause is determined and corrective action to preclude repetition is identified and tracked until it is completed and verified.

c. Specific responsibilities within the corrective action program may be delegated, but FENOC maintains responsibility for the program's effectiveness.

d. Non-conforming items are properly controlled to prevent their inadvertent test, installation, or use. They are reviewed and either accepted, rejected, repaired, or reworked.

e. Reports of conditions that are adverse to quality are analyzed to identify trends in quality performance. Significant conditions adverse to quality and significant trends are reported to the appropriate level of management.

f. Additional details concerning corrective action activities may be found in Section B.13 and the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.33 and 1.144).

7. Regulatory Commitments

a. Except where alternatives are identified, FENOC complies with the QA guidance documents listed on Table 1. If the guidance in any of these documents is in conflict with the QAPM, the guidance provided in the QAPM is the controlling guidance. Additionally, the following clarifications apply to all guidance documents listed in Table 1:

1. For modifications and nonroutine maintenance, guidance applicable to construction-like activities is applicable to comparable plant activities, except that the inspection of modifications, repairs, rework, and replacements shall be in accordance with the original design and inspection requirements or a documented approved alternative.

2. The definitions provided by Regulatory Guide 1.74 apply wherever the defined term is used in the QAPM and associated guidance documents.

3. Clarifications and alternatives to a guidance document apply wherever the guidance document is invoked.

4. In each of the ANSI Standards, other documents (e.g., other Standards, Codes, Regulations, tables, or appendices) are referenced or described. These other documents are only quality assurance program requirements if explicitly committed to in the QAPM. If not explicitly committed to, these documents are not considered as quality assurance program requirements, although they may be used as guidance.

5. ~~In cases where a regulatory guide was originally intended to apply to design or construction phase activities, or where a regulatory guide adopts a design or construction phase standard for use during the operations phase, a commitment to the regulatory guide is not intended to include recommendations that are not pertinent to operations phase activities.~~Regulatory guidance originally intended to apply to design or construction phase activities will be applied to activities during the operations phase that are comparable in nature and extent to construction phase activities.

b. The NRC is to be notified of QAPM changes in accordance with 10 CFR 50.54(a).

c. In cases where license requirements differ from the QAPM, the most stringent requirements apply.

B. PERFORMANCE/VERIFICATION

1. Methodology

a. Personnel performing work activities such as design, engineering, procurement, manufacturing, construction, installation, startup, maintenance, modification, operation, and decommissioning are responsible for achieving acceptable quality.

b. Personnel performing verification activities are responsible for verifying the achievement of acceptable quality.

c. Work is accomplished and verified using instructions, procedures, or other appropriate means that are of a detail commensurate with the activity's complexity and importance to safety.

d. Criteria that define acceptable quality are specified, and quality is verified against these criteria.

2. Design Control

a. The design control program is established and implemented to assure that the activities associated with the design of systems, components, structures, and equipment and modifications thereto, are executed in a planned, controlled, and orderly manner.

b. The program includes provisions to control design inputs, processes, outputs, changes, interfaces, records, and organizational interfaces.

c. Design inputs (e.g., performance, regulatory, quality, and quality verification requirements) are to be correctly translated into design outputs (e.g., specifications, drawings, procedures, and instructions).

d. The final design output is to relate to the design input in sufficient detail to permit verification.

e. The design process is to ensure that items and activities are selected and independently verified consistent with their importance to safety to ensure they are suitable for their intended application.

f. Changes to final designs (including field changes and modifications) and dispositions of non-conforming items to either use-as-is or repair are to be subjected to design control

measures commensurate with those applied to the original design and approved by the organization that performed the original design or a qualified designee.

g. Interface controls (internal and external between participating design organizations and across technical disciplines) for the purpose of developing, reviewing, approving, releasing, distributing, and revising design inputs and outputs are defined in procedures.

h. Design documentation and records, which provide evidence that the design and design verification process was performed in accordance with this program, shall be collected, stored, and maintained in accordance with documented procedures. This documentation includes final design documents, such as drawings and specifications, and revisions thereto and documentation which identifies the important steps, including sources of design inputs that support the final design.

i. Additional details concerning design control activities may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.64).

3. Design Verification

a. A program is established and implemented to verify the acceptability of design activities and documents for the design of items. The selection and incorporation of design inputs and design processes, outputs, and changes are verified.

b. Verification methods include, but are not limited to, design reviews, alternative calculations, and qualification testing. The extent of this verification will be a function of the importance to safety of the item, the complexity of the design, the degree of standardization, the state of the art, and the similarity with previously proven designs. Standardized or previously proven designs will be reviewed for applicability prior to use.

c. When a test program is used to verify the acceptability of a specific design feature, the test program is to demonstrate acceptable performance under conditions that simulate the most adverse design conditions that are expected to be encountered.

d. Independent design verification is to be completed before design outputs are used by other organizations for design work and before they are used to support other activities such as procurement, manufacture, or construction. When this timing cannot be achieved, the unverified portion of the design is to be identified and controlled. In all cases, the design verification is to be completed before relying on the item to perform its function.

e. Individuals or groups responsible for design reviews or other verification activities shall be identified in procedures and their authority and responsibility shall be defined and controlled. Design verification shall be performed by any competent individuals or groups other than those who performed the original design but who may be from the

same organization. The designer's immediate supervisor may perform the design verification provided: the supervisor is the only technically qualified individual capable of performing the verification, the need is individually documented and approved in advance by the supervisor's management, and the frequency and effectiveness of the supervisors use as a design verifier are independently verified to guard against abuse.

f. Design verification procedures are to be established and implemented to ensure that an appropriate verification method is used, the appropriate design parameters to be verified are chosen, the acceptance criteria are identified, the verification is satisfactorily accomplished, and the results are properly recorded.

g. Additional details concerning design verification activities may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.64).

4. Procurement Control

a. A program is established and implemented to ensure that purchased items and services are of acceptable quality.

b. The program includes provisions for evaluating prospective suppliers and selecting only qualified suppliers.

c. The program includes provisions for ensuring that qualified suppliers continue to provide acceptable products and services.

d. The program includes provisions (e.g., source verification, receipt inspection, pre-installation and post-installation tests, and certificates of conformance) for accepting purchased items and services.

e. Applicable technical, regulatory, administrative, and reporting requirements (e.g., specifications, codes, standards, tests, inspections, special processes, and 10 CFR 21) are invoked for procurement of items and services.

f. The program includes provisions for ensuring that documented evidence of an item's conformance to procurement requirements is available at the site before the item is placed in service or used unless otherwise specified in procedures.

g. The program includes provisions for ensuring that procurement, inspection, and test requirements have been satisfied before an item is placed in service or used unless otherwise specified in procedures.

h. The procurement of components, including spare and replacement parts, is subject to quality and technical requirements suitable for their intended service.

i. Appropriate controls for the selection, determination of suitability for intended use (critical characteristics), evaluation, receipt, and quality evaluation of commercial grade items are to be imposed to ensure that the items will perform satisfactorily in service.

j. Additional details concerning procurement control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.33 and 1.123).

5. Procurement Verification

a. A program is established and implemented to verify the quality of purchased items and services at intervals and to a depth consistent with the item's or service's importance to safety, complexity, and quantity and the frequency of procurement.

b. The program is executed in all phases of procurement. As necessary, this may require verification of activities of suppliers below the first tier.

c. Additional details concerning procurement verification may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.123 and 1.144).

6. Identification and Control of Items

a. A program is established and implemented to identify and control items to prevent the use of incorrect or defective items.

b. Identification of each item is maintained throughout fabrication, erection, installation, and use so that the item can be traced to its documentation. Traceability is maintained to an extent consistent with the item's importance to safety.

c. Additional details concerning identification and control of items may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.33).

7. Handling, Storage, and Shipping

a. A program is established and implemented to control the handling, storage, shipping, cleaning, and preserving of items to ensure the items maintain acceptable quality.

b. Special protective measures (e.g., containers, shock absorbers, accelerometers, inert gas atmospheres, specific moisture content levels, and temperature levels) are specified and provided when required to maintain acceptable quality.

c. Specific procedures are developed and used for cleaning, handling, storage, packaging, shipping, and preserving items when required to maintain acceptable quality.

d. Items are marked and labeled during packaging, shipping, handling, and storage to identify, maintain, and preserve the items' integrity and indicate the need for special controls.

e. Additional details concerning handling, storage, and shipping activities may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.38).

8. Test Control

a. A test control program is established and implemented to demonstrate that items will perform satisfactorily in service.

b. Criteria are defined that specify when testing is required.

c. The test control program includes, as appropriate, proof tests before installation, pre-operational tests, post-maintenance tests, post-modification tests, and operational tests.

d. Test procedures are developed that include:

1. instructions and prerequisites to perform the test,

2. use of proper test equipment,

3. acceptance criteria, and

4. mandatory inspections as required.

e. Test results are evaluated to assure that test objectives and inspection requirements have been satisfied.

f. Unacceptable test results shall be evaluated.

g. Additional details concerning test control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.33).

9. Measuring and Test Equipment Control

a. A program is established and implemented to control the calibration, maintenance, and use of measuring and test equipment. Measuring and test equipment does not include permanently installed operating equipment or test equipment used for preliminary checks where data obtained will not be used to determine acceptability or be the basis for design

or engineering evaluation. Additionally, calibration and control measures are not required for rulers, tape measures, levels and other such devices if normal commercial manufacturing practices provide adequate accuracy.

b. The types of equipment covered by the program (e.g., instruments, tools, gages, and reference and transfer standards) are defined in procedures.

c. Measuring and test equipment is calibrated at specified intervals or immediately before use on the basis of the item's required accuracy, intended use, frequency of use, and stability characteristics and other conditions affecting its performance.

d. Measuring and test equipment is labeled, tagged, or otherwise controlled to indicate its calibration status and to ensure its traceability to calibration test data.

e. Measuring and test equipment is calibrated against standards that have an accuracy of at least four times the required accuracy of the equipment being calibrated or, when this is not possible, have an accuracy that ensures the equipment being calibrated will be within the required tolerance.

f. If nationally recognized standards exist, calibration standards are to be traceable to them. Except where calibration standards with the same accuracy as the instruments being calibrated are shown to be adequate for the requirements, calibration standards are to have a greater accuracy than the standards being calibrated.

g. Measuring and test equipment found out of calibration is tagged or segregated. The acceptability shall be determined of items measured, inspected, or tested with an out-of-calibration device.

h. Additional details concerning measuring and test equipment control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.30, 1.33, 1.94, 1.116, and 1.123).

10. Inspection, Test, and Operating Status

a. The status of required inspections and tests and the operating status of items is verified before release, fabrication, receipt, installation, test, and use, as applicable. This verification is to preclude inadvertent bypassing of inspections and tests and to prevent inadvertent operation of controlled equipment.

b. The application and removal of inspection, test, and operating status indicators are controlled in accordance with procedures.

c. Additional details concerning inspection, test, and operating status control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.33).

11. Special Process Control

a. A program is established and implemented to ensure that special processes are properly controlled.

b. The criteria that establish which processes are special are described in procedures.

c. Special processes are accomplished by qualified personnel, using appropriate equipment, and procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

d. Additional details concerning special process control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.33).

12. Inspection

a. A program is established and implemented for inspections of activities in order to verify conformance to the documented instructions, procedures and drawings for accomplishing the activity. The inspection program may be implemented by or for the organization performing the activity to be inspected.

b. Provisions to ensure inspection planning is properly accomplished are to be established. Planning activities are to identify the characteristics and activities to be inspected, the inspection techniques, the acceptance criteria, and the organization responsible for performing the inspection.

c. Provisions to identify inspection hold points, beyond which work is not to proceed without the consent of the inspection organization, are to be defined.

d. Inspection results are to be documented by the inspector and reviewed by qualified personnel.

e. Unacceptable inspection results shall be evaluated and resolved in accordance with procedures.

f. Inspections are performed by qualified personnel other than those who performed or directly supervised the work being inspected. While performing the inspection activity the inspectors functionally report to the associated individual responsible for quality control or an individual responsible for materials, purchasing, and contracts as appropriate.

g. Additional details concerning inspections may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.33 and 1.58).

13. Corrective Action

a. Procedures shall provide for identification, evaluation, and resolution of conditions adverse to quality.

b. Reworked, repaired, and replacement items are to be inspected and tested in accordance with the original inspection and test requirements or specified alternatives.

c. Additional details concerning corrective action activities may be found in Section A.6 and the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.33 and 1.144).

14. Document Control

a. A program is established and implemented to control the development, review, approval, issue, use, and revision of documents.

b. The document control program shall be applied to documents that prescribe activities affecting quality of safety-related structures, systems or components. Such activities include design, procurement, material control, installation, inspection, testing, maintenance, modification, operation, refueling and decommissioning.

c. Revisions of controlled documents are reviewed for adequacy and approved for release by the same organization that originally reviewed and approved the documents or by a designated organization that is qualified and knowledgeable.

d. Controlled documents are available to and used by the person performing the activity.

e. The distribution of new and revised controlled documents is in accordance with procedures. Superseded documents are controlled to prevent inadvertent use.

f. Additional details concerning document control may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.33).

15. Records

a. A program is established and implemented to ensure that sufficient records of items and activities (e.g., design, engineering, procurement, manufacturing, construction, inspection

and test, installation, preoperation, startup, operations, maintenance, modification, decommissioning, and audits) are generated and maintained to reflect completed work.

b. The program provides provisions for the administration, receipt, storage, preservation, safekeeping, retrieval, and disposition of records.

c. Additional details concerning record requirements may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guide 1.88).

C. ASSESSMENT

1. Methodology

a. Personnel responsible for carrying out audits are maintained cognizant of day-to-day activities by the ongoing involvement in the quality assurance program requirements so that they can act in a management advisory function.

b. Organizations performing audits are to be technically and performance oriented commensurate with the activity being reviewed.

c. Personnel performing audits have no direct responsibilities in the area they are assessing.

d. Audits are accomplished using procedures, or other appropriate means that are of a detail commensurate with the activity's complexity and importance to safety.

2. Audit

a. A program of planned and periodic audits is established and implemented to confirm that activities affecting quality comply with the QAPM and that the QAPM has been implemented effectively. Audits will be conducted as required by the applicable Code of Federal Regulations, Technical Specifications, safety analysis reports, and commitments by various correspondence to the NRC. Audits will be conducted at a frequency in accordance with either Section C.2.a.1 or Section C.2.a.2 below.

1. Audit frequencies will be determined in accordance with a performance based audit scheduling program. The scheduling program, through an expert panel, uses assessment indicators to identify and schedule audits based on performance results and importance of the activity relative to safety. Potential audit subject areas are periodically assessed against appropriate performance criteria. From these reviews a determination is made in regard to the depth, scope, and scheduling of specific audits. Functional areas important to safety are assessed annually (± 25%) to identify strengths and weaknesses (if applicable) to determine the level and focus of independent oversight activities for the upcoming year. The basis for the assessment shall include the results of audits and surveillance, NRC inspections,

LERs, self-assessments, and applicable conditions reports (e.g., non-conformance and corrective action reports). Personnel changes, change/increase in functional area responsibilities, industry operating experience, and INPO evaluations will also be considered. Each area will be assigned a rating with a comparison to previous years. This assessment will be documented, reviewed, and approved by quality assurance management.

This document is considered a quality assurance record and will be available for NRC review. Audit subject areas of Section C.2.a.2 shall continue to be audited on the frequencies designated unless expert panel judgment, based on performance results, determines such an audit to be unnecessary. In such cases the expert panel basis shall be documented.

2. Audit schedules assure that the following areas are audited at the indicated frequencies, or more frequently as performance dictates. ~~Not all of the following are Appendix B audits.~~

   a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 24 months.

   b. The performance, training and qualification of the station staff at least once per 24 months.

   c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 24 months.

   d. The performance of activities required by the QAPM to meet the requirements of 10CFR50, Appendix B at least once per 24 months.

   e. The fire protection program controls and implementing procedures at least once per 24 months.

   f. The fire protection equipment and program implementation at least once per 12 months utilizing either qualified licensee personnel or an outside fire protection consultant.

   g. The fire protection equipment and program implementation at least once per 36 months utilizing a qualified outside fire protection consultant.

   h. The Radiological Environmental Monitoring Program (REMP) and radiological effluents monitoring activities and implementing procedures at least once per 24 months.

   i. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

   j. The Process Control Program and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

3. Audits shall provide an objective evaluation of quality related practices, procedures, instructions, activities, and items and a review of documents and records, as applicable.

4. Audits shall be performed in accordance with approved written procedures or checklists. Items from previous audits shall be reviewed and reaudited, as appropriate. The checklists are used as guides to the auditor.

5. Scheduling and resource allocation are based on the status and safety importance of the activity or process being assessed.

6. Scheduling is dynamic and resources are supplemented when the effectiveness of the quality assurance program is in doubt.

7. Audit reports are written and distributed to the appropriate levels of management for review. Follow-up action can be accomplished through written communication, re-audit, or other appropriate means, as deemed necessary.

8. Implementation of delegated portions of the quality assurance program is assessed.

9. Audits are conducted using predetermined acceptance criteria.

10. Additional details concerning audits may be found in the Regulatory Guides and associated Standards as committed to in Section A.7 and Table 1 (e.g., Regulatory Guides 1.33 and 1.144).

D. INDEPENDENT SAFETY REVIEW

1. Description

a. Independent safety review is performed to meet the individual unit's commitment to perform the functions described in NUREG-0737, Section I.B.1.2, "Independent Safety Engineering Group."

A. Regulatory Guide 1.8 (Revision 1) [September 1975], *Personnel Selection and Training*

1. FENOC commits to the regulatory position of this Guide with the following clarifications:

   a. Regulatory Guide 1.8 states "The RPM should have a bachelor's degree or the equivalent in a science or engineering subject including some formal training in radiation protection and at least 5 years of professional experience in applied radiation protection." It is FENOC's position that equivalent as used in this Regulatory Guide for the bachelor's degree means (a) four years of post secondary schooling in science or engineering, or (b) four years of applied experience at a nuclear facility in the area for which qualification is sought, or (c) four years of operational or technical experience or training in nuclear power, or (d) any combination of the above totaling four years. The years of experience used to meet the education requirements as allowed by this exception shall not be used to also meet the experience requirements.

   b. FENOC commits to the requirements of ANSI N18.1-1971 as modified by plant-specific Technical Specifications.

~~A.~~B. Regulatory Guide 1.30 (Revision 0) [August 1972], *Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment*

~~1. During the operations phase, testing and inspection at FENOC will be performed in accordance with the intent of this guide and the requirements of the Technical Specifications and quality assurance program.~~

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.4-1972 with the following clarifications:

   a. Section 1.1 specifies equipment to which this Standard applies. In lieu of this, requirements of this Standard shall apply to those systems and components that are within the scope of the QAPM. Each plant maintains a list of equipment subject to QAPM requirements. This Standard is also applied to other systems and components when required by approved procedures, engineering specifications, or other work controlling documents.

   b. Section 2.2 requires that evidence of compliance by the manufacturer with purchase requirements, including quality assurance requirements, be available at the site prior to applying the requirements of ANSI N45.2.4. In lieu of this requirement, installation, inspection, and testing activities of equipment lacking its quality

documentation may proceed provided that this equipment has been identified and released in accordance with non-conforming material procedures and that all required quality documentation has been received and accepted prior to the item being placed in service.

c. Section 3 requires that records of protective measures maintained during storage for conformance to storage requirements be checked to verify that items are in satisfactory condition for installation. This check shall be made only if equipment requires special storage or handling as specified in procurement documents.

d. Sections 5.2 and 6.2 list the tests which are to be conducted during construction and post-construction activities. In lieu of these tests, FENOC shall conduct only those tests necessary to verify that work activities specified by work controlling documents have been satisfactorily accomplished during maintenance or modification activities. The requirements of Sections 5.2 and 6.2 of ANSI N45.2.4 shall be used as guidelines in determining these testing requirements.

e. Section 6.2.1 states in part that "Items requiring calibration shall be tagged or labeled on completion indicating date of calibration and identity of person that performed the calibration." In lieu of this requirement, FENOC may alternatively implement programs that require the equipment to be suitably marked to indicate the date of the next calibration and the identity of the person that performed the calibration.

~~B.~~C. Regulatory Guide 1.33 (Revision 2) [February 1978], *Quality Assurance Program Requirements (Operations)*

1. FENOC commits to the regulatory position of this Guide with the following alternatives:

   a. Regulatory Position C.3 specifies review of proposed license amendments by the independent review body prior to submittal to the NRC. As an alternative, a committee that is part of the onsite operating organization may perform this review.

   b. Regulatory Position C.4 specifies audit frequencies for several audit topics. QAPM Section C.2 (Audit) describes alternatives to these frequencies.

2. FENOC commits to the requirements of ANSI N18.7-1976/ANS 3.2 with the following clarifications:

   a. Section 1 requires that this Standard "apply to all activities affecting the safety-related functions of nuclear power plant structures, systems, and components."

FENOC shall apply the requirements of this Standard to those structures, systems, and components identified as safety-related in the respective plant's USAR.

b. Section 5.1 states in part that "a summary document should be compiled by each owner organization to identify the sources, to index such sources to the requirements of this Standard, and to provide a consolidated base for the description of the program." In lieu of this requirement, a method of cross-referencing these requirements to the implementing procedures will be maintained.

c. Section 5.2.2 requires that "temporary changes which clearly do not change the intent of the approved procedure shall, as a minimum, be approved by two members of the plant staff knowledgeable in the areas affected by the procedure. At least one of these shall be the supervisor in charge of the shift and hold a senior operating license on the unit affected." Such changes shall be documented and if appropriate, incorporated into the next revision of the affected procedure. -In lieu of these requirements, FENOC commits to ~~the requirements as delineated in the site's USAR, Technical Specifications or technical requirements manual.~~ the following:

1) Temporary changes to procedures which do not change the intent of the approved procedure shall be approved for implementation by two members of the plant management staff, at least one of whom holds a Senior Reactor Operating License for the unit affected. The temporary procedures shall be approved by the original approval authority within 14 days. For changes to procedures which may involve a change in intent of the procedure, the original approval authority shall approve the change prior to implementation. OR

2) Temporary changes to procedures will be approved by two knowledgeable members of the plant staff prior to implementation. At least one of these persons will be a member of supervision. If the change affects operations procedures, at least one of these persons will hold a senior reactor operator license for the unit affected. Prior to implementation, the OSC (PORC) shall review and recommend approval of temporary changes to procedures which require a 10CFR50.59 safety evaluation. Within 14 days of implementation, temporary changes will be reviewed by an independent qualified reviewer and approved by the Responsible Discipline Manager or his designee.

d. Section 5.2.6 requires that a log be maintained to identify the current status of temporary modifications such as bypass lines, electrical jumpers, lifted electrical leads, and temporary trip point settings. FENOC takes exception to this requirement when the installation and removal of such temporary modifications is specifically addressed in approved procedures. These procedures ensure that the circuitry is returned to its original configuration when the operation is completed.

e. Section 5.2.7 – Since certain emergency situations could arise which might prevent preplanning activities, FENOC complies with an alternative to the first sentence in the second paragraph as follows: "Except under emergency or abnormal operating conditions where immediate actions are required to protect the health and safety of the public, to protect equipment or personnel, or to prevent the deterioration of plant conditions to a possibly unsafe or unstable level, maintenance or modification of equipment shall be preplanned and performed in accordance with approved procedures. When written procedures would be required and are not used, the activities that are accomplished are documented after-the-fact and receive the same degree of reviews as if they had been preplanned."

f. Section 5.2.15 contains a requirement for biennial review of plant procedures. In lieu of this requirement, FENOC may use one of the following methods as alternatives:

1) Implement process controls that ensure procedures are reviewed for possible revision upon identification of new or revised source material potentially affecting the intent of procedures.

2) Implement process controls related to procedure review, a maximum six year review period and biennial audits of operating organizations that include a review of their procedures to assure that controls result in timely procedure revision in response to operations experience deficiencies and procedure deficiencies identified by users.

~~C.~~D. Regulatory Guide 1.37 (Revision 0) [March 1973], *Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide with the following clarifications:

a. Regulatory Position C.3 requires that water quality for final flushes of fluid systems and associated components be at least equivalent to the quality required for normal operation. This requirement is not applied to dissolved oxygen or nitrogen nor does it infer that additives normally in the system water shall be added to the flush water.

b. Regulatory Position C.4 requires that chemical components that could contribute to intergranular cracking or stress corrosion cracking should not be used with austenitic stainless steel and nickel-based alloys. It is FENOC's position that materials such as inks, temperature indicating crayons, labels, wrapping materials (other than polyethylene), water soluble materials, desiccants, lubricants, and NDE penetrant materials and couplants, which contact stainless steel or nickel-based alloy material surfaces contain no more than trace amounts of lead, zinc, copper, or lower melting

alloys or compounds. Maximum allowable levels of water leachable chloride ions, total halogens and sulfur compounds shall be defined and imposed on the aforementioned materials. These materials will be controlled through administrative procedures that are, in part, designed to minimize their effects on intergranular cracking or stress corrosion cracking.

2. FENOC commits to the requirements of ANSI N45.2.1-1973 with the following clarifications:

   a. During maintenance and modification activities, FENOC shall control the opening of clean systems and shall conduct inspections to verify that affected system cleanliness levels shall not be adversely affected by the maintenance or modification activity. When system cleanliness is affected, specific cleaning procedures which incorporate the applicable portions of this Standard shall be developed and implemented to maintain system cleanliness.

   b. Section 2.4 requires that personnel who perform inspection, examination or testing activities required by this Standard be qualified in accordance with ANSI N45.2.6. In lieu of this, personnel who perform cleanliness inspections may alternatively be qualified in accordance with Regulatory Guide 1.8.

~~D.~~E. Regulatory Guide 1.38 (Revision 2) [May 1977], *Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.2-1978 with the following clarifications:

   a. Sections 3 and 4 specify a four level classification system for the packaging and shipping of items. In lieu of these requirements, commercial grade items shall be packaged and shipped in accordance with standard commercial practices.

   b. Section 5.2.1 requires preliminary visual inspection or examination for shipping damage to be performed prior to unloading. In lieu of this requirement, visual inspection shall be performed during unloading and unpacking.

   c. Section 5.5 provides for "rework" and "use-as-is" dispositions for nonconforming items. As an alternative, the "repair" disposition (as defined by ANSI N45.2.10-1973) may also be used.

d. Section 6.5 requires that items released from storage and placed in their final locations within the power plant be inspected and cared for in accordance with the requirements of Section 6 of this Standard and other applicable Standards. In lieu of this requirement, FENOC shall, whenever feasible, store items within their appropriate storage area and move the equipment to the plant areas for staging only in sufficient time to support its installation. Within the plant, the equipment shall be staged at locations which provide equivalent environmental conditions under which it is designed to operate. Materials placed in staging areas shall be stored in accordance with the applicable requirements of Paragraphs 6.1, 6.3 and 6.4.2 of ANSI N45.2.2.

e. Various Sections of ANSI N45.2.2 address the use of non-halegenated materials when in contact with austenitic stainless steel or nickel-based alloys. The exceptions applicable to Regulatory Guide 1.37 regarding this subject also apply to ANSI N45.2.2.

f. Section A.3.4.2 addresses inert gas blankets. There may be cases involving large or complex shapes for which an inert or dry air purge flow is provided rather than static gas blankets in order to provide adequate protection due to difficulty of providing a leak-proof barrier. In these cases, a positive pressure purge flow may be used as an alternative to a leak-proof barrier.

~~E.~~F. Regulatory Guide 1.39 (Revision 2) [September 1977], *Housekeeping Requirements for Water-Cooled Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.3-1973 with the following alternative.

   a. The ANSI five level zone designation system may not be utilized, but the intent of the standard will be met for the areas of housekeeping, plant and personnel safety, and fire protection.

~~F.~~G. Regulatory Guide 1.58 (Revision 1) [September 1980], *Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel*

1. FENOC commits to the regulatory position of this Guide with the following clarifications:

   a. The guidance of this Regulatory Guide shall be followed as it pertains to the qualification of personnel who verify conformance of work activities to quality requirements.

b. Personnel will not be certified as stated in this Guide in the following areas:

1) Individuals that handle test results or perform document control activities.

2) Quality assurance and staff personnel responsible for the review of documents for clarity and completeness.

3) Test personnel utilizing gas test methods for information or data collection activities (this includes those personnel performing local leak rate testing (LLRT) as stated in 10CFR50 Appendix J). The qualifications of these personnel shall conform to the requirements of Regulatory Guide 1.8.

4) Plant operation personnel concerned with day-to-day operation, maintenance, and certain technical services (the qualifications of these personnel shall conform to the requirements of Regulatory Guide 1.8).

c. Regulatory Position C.2 indicates that SNT-TC-1A-1975 is to be used for the qualification of nondestructive examination (NDE) personnel who apply various NDE methods. It also indicates that personnel performing nondestructive examinations required by Section III and Section XI of the ASME Code should be qualified to SNT-TC-1A-1975 as well as additional provision of the Code. For the qualification of NDE personnel, FENOC commits to the ASME Section XI requirements specified within the applicable code year edition(s) as defined by 10CFR50.55a. This alternative may be applied regardless of whether examinations are of a type required by the Code.

2. FENOC commits to the requirements of ANSI N45.2.6-1978 as modified by the commitments to Regulatory Guide 1.58 with the following clarifications:

a. Section 2.5 of this Standard discusses special physical characteristics. FENOC commits to the following: Examinations to verify that personnel have the required physical characteristics will be scheduled on an annual basis with a maximum allowable extension of 90 days.

b. Section 3.5 of this Standard discusses education and experience. FENOC commits to the following: The initial qualifications of individuals to Level I, II, or III will generally be to the education and experience recommendations in the Standard. However, in certain instances as determined by appropriate management, qualifications may be alternatively determined through test results and/or demonstration of capabilities. For Level I, FENOC will also accept a four year college degree plus one month of related experience or equivalent inspection,

examination or testing activities. Individual requalification will meet or exceed the recommendation of this Standard.

~~G.~~H. Regulatory Guide 1.64 (Revision 2) [June 1976], *Quality Assurance Requirements for the Design of Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide with the following clarifications:

   a. Regulatory Position C.2(1) addresses the use of a supervisor in design verification. If, in exceptional circumstances, the supervisor is the only technically qualified individual available, the design verification or checking shall be conducted by the supervisor with the following provisions:

      1) The other requirements of Regulatory Position C.2 of this Guide shall be met.

      2) The justification shall be individually documented and approved by the next level of supervision.

      3) Quality assurance audits shall include review of frequency and effectiveness of the use of the immediate supervisor to assure that this provision is used only in exceptional circumstances.

   b. An individual who contributed to a given design may participate in a group verification of that design provided that the individual who contributed to the design does not (1) verify his contribution to the design, or (2) serve as chairman or leader of the group verification activity.

2. FENOC commits to the requirements of ANSI N45.2.11-1974 with the clarifications as noted above for the use of an immediate supervisor for design verification activities and conduct of group verification activities.

~~H.~~I. Regulatory Guide 1.74 (Revision 0) [February 1974], *Quality Assurance Terms and Definitions*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.10-1973.

~~I.~~J. Regulatory Guide 1.88 (Revision 2) [October 1976], *Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.9-1974 with the following alternatives:

   a. Section 5.6 addresses records storage facilities. In lieu of this, the design and construction of quality assurance record storage facilities will follow the guidance of ANSI/ASME NQA-1-1983, Supplement 17S-1, Section 4.4. When temporary storage of records is required, the guidance of ASME NQA-1-1989, Supplement 17S-1, Section 4.4.3 will be followed. For storage of special processed records (such as radiographs and microfilm), humidity and temperature controls shall be provided so as to maintain an environmental condition as prescribed in Paragraph 6.1.1 of ANSI PH 1.43-1979 (Also required by Section 5.4).

   b. Appendix A of ANSI N45.2.9, requires that records of measuring and test equipment calibration be maintained "until recalibration." This implies that the full storage requirements of this Standard apply until the equipment is recalibrated. In lieu of this requirement, FENOC may store measuring and test equipment calibration records in one-hour fire rated containers. This exception does not apply to records of calibration required by the Technical Specifications.

~~J.~~K. Regulatory Guide 1.94 (Revision 1) [April 1976], *Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.5-1974 with the following clarification:

   a. Section 2.2 requires that installation, inspection, and test procedures be kept current with the latest information. This Standard was written to address requirements associated with construction phase activities. However, during the operations phase, activities associated with installation, inspection, and testing of structural concrete and structural steel are very minor in frequency and extent. Consequently, procedures for these activities shall only be reviewed or updated prior to commencing the activity. The procedures for structural concrete and structural steel installation, inspection, and testing activities will be developed using the provisions of ANSI N45.2.5 – 1974.

   b. Alternatives to this Standard are taken with respect to frequency of calibration of impact wrenches and bolt projection criteria. Impact and torque wrenches shall be checked at least once daily per shift, and at least one full thread of all bolts shall

project beyond the nut of all tightened connections. These criteria comply with the recommendations of the Research Council on Riveted and Bolted Structural Joints.

~~K.~~L. Regulatory Guide 1.116 (Revision 0) [May 1977], *Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.8-1975 with the following clarifications:

   a. Sections 2.4 and 2.6 require that procedures define system restoration requirements as needed to prevent contamination after cleanliness class is achieved in accordance with commitments to ANSI N45.2.1 and ANSI N45.2.3.

   b. Section 2.9 requires that evidence of compliance by the manufacturer with purchase requirements, including quality assurance requirements, be available at the site prior to applying the requirements of this Standard. In lieu of this requirement, section B.4 (Procurement Control) of this manual describes the controls for equipment lacking quality documentation.

   c. Section 4.5.1 provides requirements for the cleaning, flushing, and conditioning of installed systems. FENOC's position on Regulatory Guide 1.37 and ANSI N45.2.1 also apply to this Section and take precedence over the requirements of ANSI N45.2.8 when conflicts exist.

~~L.~~M. Regulatory Guide 1.123 (Revision 1) [July 1977], *Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.13-1976 with the following clarifications:

   a. Section 4 provides for the selection of procurement sources. For "commercial grade" items and for non-safety related items within the scope of the Quality Assurance Program for which there are no quality assurance program or quality documentation requirements, the requirements of this Section need not be adhered to. However, the procurement documents shall specify requirements specific to the item being procured, sufficient to provide adequate certification or other records to ensure that items and activities meet the specified requirements.

b. Section 8.2 provides requirements for the control of nonconformances. Suppliers qualified by FENOC as design agents in accordance with Regulatory Guides 1.64 and 1.123 may be permitted under specific contractual provisions to disposition nonconformances as "use-as-is" or "repair" on behalf of FENOC. All nonconformances dispositioned "use-as-is" or "repair" by suppliers qualified by FENOC as design agents on behalf of FENOC are required to be submitted to FENOC for engineering approval at the time equipment is received on site. If FENOC determines that a disposition has been incorrectly made, a nonconformance report is generated on site to document the problem and effect resolution.

c. Section 10.2.d is interpreted as follows: The person attesting to a certificate shall be an authorized and responsible employee of the supplier and shall be identified by the supplier.

~~M.~~N. Regulatory Guide 1.144 (Revision 1) [September 1980], *Auditing of Quality Assurance Programs for Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.12-1977 with the following clarification:

   a. Section 4.5.1 of this Standard discusses follow-up and corrective actions. FENOC may utilize the provisions of the corrective action program outlined in Section A.6 instead of these requirements, as long as the appropriate time limits are applied to significant conditions adverse to quality. Also, no additional documentation is necessary if needed corrective actions are taken and verified prior to audit report issuance.

   b. Sections 4.3.1 and 4.3.3 of this Standard discuss pre-audit and post-audit conferences. Pre-audit and post-audit conferences may be fulfilled by a variety of communications, such as telephone conversation.

   c. Section 4.3.1 and 4.3.3 of this Standard discuss pre-audit and post-audit conferences. Pre-audit and post-audit conferences are only held when deemed necessary by quality assurance or when requested by the audited organization.

   d. Section 4.4 discusses audit reporting. Audit reports shall be issued within thirty working days after the last day of the audit. The last day of an audit shall be considered to be the day of the post-audit conference. If a post-audit conference is not held because it was deemed unnecessary, the last day of the audit shall be

considered to be the date the post-audit conference was deemed unnecessary as documented in the audit report.

~~N.~~O. Regulatory Guide 1.146 (Revision 0) [August 1980], *Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants*

1. FENOC commits to the regulatory position of this Guide.

2. FENOC commits to the requirements of ANSI N45.2.23-1978 with the following alternatives.

   a. Section 2.3.1.3 discusses other credentials of professional competence. Holders of NRC issued Reactor Operator/Senior Reactor Operator Licenses comply with the requirements of this section and may be awarded two credits.

   b. Section 2.3.4 discusses audit participation. Prospective lead auditors shall demonstrate their ability to effectively implement the audit process and lead an audit team. They shall have participated in at least one nuclear audit within the year preceding the individual's effective date of qualification. Upon successful demonstration of the ability to effectively lead audits, licensee management may designate a prospective lead auditor as a lead auditor.

ATTACHMENT 3

BVPS-2 UFSAR Excerpts

TABLE 1.8-1 (Cont)

organic and inorganic species if 2-inch charcoal bed depth is provided; 99 percent if 4 or more inches of charcoal bed depth is provided) since these represent more realistic values.

Paragraph C.1.d specifies that the analysis should be performed assuming 10% of the total radioactive iodine in the rods of the time of the accident. However, the iodine percentages used are 12% I-131 and 10% of the other iodine nuclides. This is in keeping with NUREG-5009, as referenced by the USNRC in the safety evaluation report for license amendment 12.

RG No. 1.26, Rev. 3
UFSAR Reference Section 3.2.2

QUALITY GROUP CLASSIFICATIONS AND STANDARDS FOR WATER-, STEAM-, AND RADIOACTIVE-WASTE-CONTAINING COMPONENTS OF NUCLEAR POWER PLANTS (FEBRUARY 1976)

Quality group classifications and standards for water-, steam-, and radioactive-waste-containing components of Beaver Valley Power Station - Unit 2 meet the intent of Regulatory Guide 1.26 with the following alternatives:

1. The safety class terminology of ANSI N18.2 and ANSI 18.2a-1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and Non-nuclear Safety (NNS) Class are used instead of Quality Groups A, B, C, and D, respectively, and are consistent with present nuclear industry practice.

2. Paragraph NB-7153 of the ASME Section III Code requires that there be no valves between a code safety valve and its relief point unless special interlocks prevent shutoff without other protection capacity. Therefore, as an alternative to Paragraphs C.1.e and C.2.c, a single safety valve designed, manufactured, and tested in accordance with ASME III Division 1 is considered acceptable as the boundary between the reactor coolant pressure boundary and a lower safety class or NNS class line.

3. Portions of the emergency diesel generator cooling water system, considered by the vendor to be parts of the engine (as distinguished from auxiliary support systems), were built to the manufacturer's standards rather than ASME III. These are identified in Table 3.2-1 and Section 9.5.5. The components used are of high quality, proven by experience, and were designed, fabricated, erected, and tested under the vendor's Quality Assurance Program which meets the requirements of 10CFR50, Appendix B. Similar equipment has been accepted by the NRC for other nuclear power plant applications.

TABLE 1.8-1 (Cont)

4. Regarding Regulatory Positions C.1 and C.2, all instrument tubing, classified as Safety Class 2 or 3, are designed to ASME Section III rules and installed in accordance with the BVPS-2 Quality Assurance Program for safety-related equipment.

RG No. 1.27, Rev. 2
UFSAR Reference Sections 2.4.11.6, 9.2.5

ULTIMATE HEAT SINK FOR NUCLEAR POWER PLANTS (JANUARY 1976)

The ultimate heat sink for Beaver Valley Power Station - Unit 2 follows the guidance of this regulatory guide.

RG No. 1.28, Rev. 2
UFSAR Reference Sections 17.1.2, 17.2

QUALITY ASSURANCE PROGRAM REQUIREMENTS (DESIGN AND CONSTRUCTION) (FEBRUARY 1979)

This regulatory guide does not apply to the Beaver Valley Power Station - Unit 2 (BVPS-2) Quality Assurance Program since it is applicable to construction permit applicants docketed after October 1979. BVPS-2, docketed October 20, 1972, meets the requirements of Appendix B to 10 CFR 50 with the BVPS-2 Design and Construction Quality Assurance Program submitted and approved through Appendix A of the BVPS-2 PSAR. Regulatory Guide 1.28 does not apply to the BVPS-2 Quality Assurance Program for plant operations since it is applicable only to the plant design and construction phase.

RG No. 1.29, Rev. 3
UFSAR Reference Section 3.2.1

SEISMIC DESIGN CLASSIFICATION (SEPTEMBER 1978)

The seismic design classification of structures, systems, and components at Beaver Valley Power Station Unit 2 follows the guidance of Regulatory Guide 1.29 with the following clarifications:

Components within the NSSS vendor scope of supply which are placed in Safety Class 3 per ANS 18.2, Paragraphs 2.2.3 (1), (3), or (4) may be classified Seismic Category II if failure during or following an ANS Condition II event would result in consequences no more severe than allowed for an ANS Condition III event.

TABLE 1.8-1 (Cont)

For the Balance of Plant, each component which is required to mitigate the consequences of an accident, as defined in ANSI N18.2, shall be classified Seismic Category I. In addition, all components classified as Safety Class 1, 2, or 3 shall be designated Seismic Category I. Seismic Category I components, structures, and systems shall be designed to remain functional in the event of the safe shutdown earthquake (SSE). All Seismic Category I components are designed and constructed to Quality Assurance (QA) Category I requirements.

Portions of structures, systems, or components whose continued function after an SSE are not required, but whose failure could reduce the functioning of other safety-related structures, systems, or components shall be designated Seismic Category II. These structures, systems, or components shall either be seismically designed, located to preclude interactions, further restrained, structurally upgraded, or proven incapable of affecting safety.

Seismic Category I design requirements shall extend to the first seismic restraint beyond the seismic boundary and shall include the interface portion of the boundary itself (that is, for piping systems, the isolation valve at a boundary between Seismic Category I and nonseismic portions shall be designated Seismic Category I. The piping up to and including the first seismic restraint beyond the valve shall be designed to Seismic

TABLE 1.8-1 (Cont)

5. Compliance with single failure criteria will be verified based on a collective analysis of both the protective system and the final actuation devices or actuators.

RG No. 1.54, Rev. 0
UFSAR Reference Sections 6.1.2, 17.1

QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS (JUNE 1973)

Quality assurance requirements for protective coatings at BVPS-2 meet the intent of this regulatory guide with the following clarification and alternatives:

For the balance-of-plant, ANSI N101.4-1972 requirements for documentation are applied as follows to equipment located in the containment.

For large surface area components, the documents are submitted by the vendors as required by ANSI N101.4-1972. These components include such items as the polar crane, structural steel, concrete, ductwork, uninsulated pipe, neutron shield tank, exteriors of uninsulated tanks and vessels, major equipment supports, and the containment liner.

For manufactured equipment such as pumps, motors, pipe hangers, and supports, the documentation required by ANSI N101.4-1972 is maintained in the seller's files for the complete duration of the contract warranty-guarantee period. A certificate of compliance signed by responsible management personnel is furnished by the seller.

For balance of plant, in lieu of the inspection defined in Section 6.2.4 of ANSI 101.4-1972, inspection is performed in accordance with ANSI N5.12-1974, Section 10, "Inspection for Shop and Field Work."

For nuclear steam supply equipment located in the containment, the following acceptable alternate method is employed.

For large surface area components, Westinghouse specifies stringent requirements through the use of a painting specification which includes the use of specific coating systems qualified to ANSI N101.2 and certifications of compliance from the vendors. The vendor's implementation of the specification requirements is monitored during the quality assurance surveillance activities. These components include the reactor

TABLE 1.8-1 (Cont)

coolant system supports, reactor coolant pumps, accumulator tanks, and the manipulator crane.

For intermediate surface area components, Westinghouse employs another specification which also includes the use of coating systems which are qualified to ANSI N101.2. The vendor's compliance with the requirements is also checked during quality assurance surveillance activities. These components include the seismic platform and tie rods, reactor internals lifting rig, head lifting rig, and electrical cabinets.

For both the nuclear steam supply system and the balance-of-plant, Regulatory Guide 1.54 guidelines are not invoked for items such as valve bodies, handwheels, certain electrical cabinets and control panels, loudspeakers, emergency light cases, and small instruments. The total surface area of these items is very small in comparison with the total surface area for which the guidelines are imposed.

The guidelines of this regulatory guide are not applied to routine touch-up work.

No special QA requirements are imposed for the painting of surfaces that will be insulated.

In general, stainless steel and corrosion-resistant alloys are not painted.

RG No. 1.55, Rev. 0
UFSAR Reference Section 3.8.1.6.1

CONCRETE PLACEMENT IN CATEGORY I STRUCTURES (JUNE 1973)

Regulatory Guide 1.55 was withdrawn (June 1981) and has been superseded by Regulatory Guide 1.136, Rev. 2, June 1981, (Materials, Construction and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments")). However, since a significant portion of Beaver Valley Power Station - Unit 2 (BVPS-2) design and construction was completed prior to the withdrawal of this regulatory guide, concrete placement in Category I structures at BVPS-2 meets or exceeds the intent of Regulatory Guide 1.55 with the following alternatives:

1. Shop detail drawings for the reactor containment mat, shell, dome, and internals are checked by the designer. All other reinforcing shop details are checked by engineers at the job site.

TABLE 1.8-1 (Cont)

2. Constituents and proportions for design mixes to be used for mass concrete are selected to minimize the effects of shrinkage and heat of hydration. The slump used for mass concrete is 3 inches, the slump used in normal concrete is 4 inches, and a slump of 5 inches is allowed in congested areas of heavily reinforced structures and electrical duct lines to permit placing concrete.

3. Curing and protection of freshly deposited concrete conforms to ACI-301, Chapter 12, except that curing compounds are not used on surfaces to which additional concrete is to be bonded, and where wood and/or metal forms are used and remain in place for curing, the forms are kept wet as required to prevent their opening at the joints and drying out of the concrete.

4. The ACI and ASTM specifications are supplemented as necessary with mandatory requirements relating to types and strengths of concrete, minimum concrete densities, proportioning of ingredients, reinforcing steel requirements, joint treatments, and testing agency requirements.

RG No. 1.56, Rev. 1

MAINTENANCE OF WATER PURITY IN BOILING WATER REACTORS (JULY 1978)

This regulatory guide is not applicable to Beaver Valley Power Station - Unit 2.

RG No. 1.57, Rev. 0
UFSAR Reference Section 3.8.2

DESIGN LIMITS AND LOADING COMBINATIONS FOR METAL PRIMARY REACTOR CONTAINMENT SYSTEM COMPONENTS (JUNE 1973)

The design limits and loading combinations for the Beaver Valley Power Station - Unit 2 metal primary reactor containment system components meet the intent of Regulatory Guide 1.57 with the following alternatives which apply only to those portions not backed by concrete:

1. The applicable edition of the ASME Boiler and Pressure Vessel Code for affected ASME III components is identified in the ASME Code Baseline Document.

2. The primary stresses, based on elastic analysis, meet the following limits in lieu of the limits specified in paragraph C.1.b(2):

TABLE 1.8-1 (Cont)

RG No. 1.78, Rev. 0
UFSAR Reference Sections 2.2.3, 6.4, 9.4

ASSUMPTIONS FOR EVALUATING THE HABITABILITY OF A NUCLEAR POWER PLANT CONTROL ROOM DURING A POSTULATED HAZARDOUS CHEMICAL RELEASE (JUNE 1974)

Assumptions for evaluating the habitability of the Beaver Valley Power Station - Unit 2 (BVPS-2) control room during a postulated hazardous chemical release meet the intent of this regulatory guide with the following clarifications and alternative:

1. Of the various evaluation methods available, BVPS-2 evaluation has been performed by the methodology outlined in NUREG-0570, published in June 1979, which is similar to that presented in Appendix B of Regulatory Guide 1.78 but at a much greater level of detail and refinement.

2. Protection of the control room during a chlorine release is addressed in the BVPS-2 position on Regulatory Guide 1.95.

Paragraph C.9

The existing control room emergency bottled air pressurization system, which will be used to pressurize both the BVPS-1 and BVPS-2 control rooms, will provide a minimum positive pressure differential of 1/8 inch water gauge in the control room relative to the space surrounding the control room.

ATTACHMENT 4

BVPS-1 UFSAR Excerpts

Probabilities have been established by the use of general failure data based on continuous operation. Specific probability analyses will be provided on a plant basis at the request of the commission.

3. "The equipment can routinely be tested when the reactor is shut down."

In all the cases discussed above, it is only the device function that is not tested. The logic associated with the devices has the capability for testing at power.

Refer to Sections 7.2 and 7.3 for further discussion.

1.3.3.23 Onsite Meteorological Programs (Safety Guide 23)

The BVPS-1 onsite meteorological program complies with Regulatory Guide 1.23 as described in Section 2.2.3.

1.3.3.24 Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)

The assumptions used for evaluating the potential radiological consequences of radioactive gas storage tank ruptures are provided in Section 14.2.3.

1.3.3.25 Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)

The assumptions used for evaluating the potential radiological consequences of a fuel handling accident are provided in Section 14.2.1.

1.3.3.26 Quality Group Classifications and Standards (Safety Guide 26)

Components for BVPS-1 were classified by quality assurance categories, as discussed in Appendix A.1.

Compliance with GDC 1 is discussed in Appendix 1A.1. Design and fabrication criteria for the Engineered Safety Feature (ESF) equipment is covered in Section 6.2. The codes and standards applicable to other systems and components are discussed within the respective sections.

1.3.3.27 Ultimate Heat Sink (Safety Guide 27)

The ultimate heat sink of BVPS-1 is the Ohio River. The river is the water source of the cooling water system that removes residual heat after reactor shutdown and following an accident.

2. Site related events or accidental phenomena could not cause diversion or loss of water for the same reasons as discussed in Section 2.1.7.

3. Since the ultimate heat sink does not depend on man-made features there is no single failure of a man-made structure that can cause a loss of cooling water.

1.3.3.28 Quality Assurance Program Requirements (Design, Construction and Operation) (Safety Guide 28)

The Duquesne Light Company Quality Assurance Program during the design and construction phase is described in Appendix A.2.1 and the operational phase is described in Section A.2.2. This program was intended to fulfill the intent of Appendix B to 10CFR50.

The Stone & Webster Quality Assurance Program is described in Appendix A.3. This program implemented the intent of 10CFR50, Appendix B.

The Westinghouse Quality Assurance Plan described in Appendix A.4 for safety related NSSS equipment complied with the requirements of ANSI N45.2.[8] The requirements provided therein apply to the design and fabrication of safety related equipment, and therefore, satisfies Safety Guide 28.

1.3.3.29 Seismic Design Classification (Safety Guide 29)

The seismic design of the BVPS-1 structures and components is discussed in Appendix B. Seismic Category I components, systems and structures are listed in Table B.1-1. The NSSS fluid systems component seismic category list is given in Table B.3-1.

The terminology Operational Basis Earthquake (OBE) and Design Basis Earthquake (DBE) is considered comparable to the terms 1/2 Safe Shutdown Earthquake (1/2 SSE) and Safe Shutdown Earthquake (SSE).

1.3.3.30 Quality Assurance Requirements for the Installation Inspection and Testing of Instrumentation and Electric Equipment (Safety Guide 30)

Quality assurance requirements for the installation, inspection and testing of instrumentation and electric equipment is, to the greatest extent possible, in accordance with Safety Guide 30. The quality assurance program is in Appendix A.

be required to determine the acceptability of the welds. The sample size shall be 10 percent of the welds in the system or component. If any of these weld samples are defective, that is, fail to pass bend tests as prescribed by ASME Code, Section IX, all remaining welds shall be sampled and all defective welds shall be removed and replaced."

1.3.3.32 Use of IEEE STD-308-1971 "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations" (Safety Guide 32)

Class 1E electric systems, to the greatest extent possible, comply with Safety Guide 32.

Availability of offsite power is discussed in Appendix 1A.17.

The capacity of each battery charger supply is based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery to the fully charged state, irrespective of the status of the plant during which these demands occur.

1.3.3.33 Quality Assurance Program Requirements (Operation) (Safety Guide 33)

BVPS-1 has formed a Quality Assurance Department. This department is responsible for the administration of the operational quality assurance program.

The BVPS-1 Quality Assurance Manual has been revised to incorporate quality assurance for operations. This program complies with AEC Safety Guide 33. ANSI N45.2 and ANSI N18.7[9] (previously ANS 3.2) requirements are referenced within Safety Guide 33.

BVPS-1 Quality Control is responsible for the preparation of the quality control procedures necessary to comply with Safety Guide 33.

REGULATORY GUIDES

1.3.4 Guidelines Used for the Operations Quality Assurance Program

1.3.4.1 Regulatory Guides

REGULATORY GUIDE 1.33, NOVEMBER 3, 1972: QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATIONS)

The Operations Quality Assurance Program requirements follow the guidance of Regulatory Guide 1.33, November 3, 1972 [including referenced standards ANSI N45.2, 1971 and ANSI N18.7, 1972 (formerly ANS 3.2)].

Appendix A of Regulatory Guide 1.33, (Revision 2, February 1978), is used as guidance to ensure minimum procedural coverage for plant activities.

The biennial review of safety related plant procedures described in ANSI N18.7 will be replaced by programmatic controls related to procedure review found in plant administrative procedures, and a maximum six year procedure review period. Biennial audits of operating organizations will include a review of their procedures to provide additional assurance that existing programmatic controls are resulting in the timely revision of their procedures in response to operations experience deficiencies and procedure deficiencies identified by users.

REGULATORY GUIDE 1.37, MARCH 16, 1973: QUALITY ASSURANCE REQUIREMENTS FOR CLEANING OF FLUID SYSTEMS AND ASSOCIATED COMPONENTS OF WATER-COOLED NUCLEAR POWER PLANTS

The Operations Quality Assurance Program requirements follow the guidance of Regulatory Guide 1.37. Procedures and/or specifications were developed prior to, and implemented concurrent with the start of the operations phase.

REGULATORY GUIDE 1.38, MARCH 16, 1973: QUALITY ASSURANCE REQUIREMENTS FOR PACKAGING, SHIPPING, RECEIVING, STORAGE, AND HANDLING OF ITEMS FOR WATER-COOLED NUCLEAR POWER PLANTS

The Operations Quality Assurance Program requirements follow the guidance of Regulatory Guide 1.38. Procedures and/or specifications were developed prior to, and implemented concurrent with the start of the operations phase.

REGULATORY GUIDE 1.39, MARCH 16, 1973: HOUSEKEEPING REQUIREMENTS FOR WATER-COOLED NUCLEAR POWER PLANTS

The Operations Quality Assurance Program requirements follow the guidance of Regulatory Guide 1.39. Procedures and/or specifications were developed prior to, and implemented concurrent with the start of the operations phase.

REGULATORY GUIDE 1.54, JUNE, 1973: QUALITY ASSURANCE REQUIREMENTS FOR PROTECTIVE COATINGS APPLIED TO WATER-COOLED NUCLEAR POWER PLANTS

The Operations Quality Assurance Program requirements follow the guidance of Regulatory Guide 1.54. Procedures and/or specifications were developed prior to, and implemented concurrent with the start of the operations phase.

Provisions for detection have been furnished in the chlorination building. Chlorine gas can be detected at concentrations as low as 1.0 ppm. Alarms have been set at 1.0 ppm. Olfactory recognition is approximately 3.5 ppm. The maximum concentration for continuous human habitation is 50 ppm, at which level respiratory trouble is experienced. The control room cannot attain levels of chlorine resulting from a storage accident that would make it uninhabitable or cause its evacuation.

Three seismically qualified chlorine detectors are installed to sample the control room air intake. These detectors will alarm and automatically initiate control room isolation and the emergency bottled air supply upon sensing a chlorine concentration in excess of a setpoint which will be set at 5 ppm or less.

Following the first hour, self-contained breathing apparatus units and sufficient reserve air cylinders are available to support the minimum control room shift composition for at least five additional hours. This satisfies Regulatory Guides 1.78 and 1.95. Sufficient additional units are provided to support the members of the emergency squad stationed outside the control room for one hour, after which these personnel would move away from the area affected by the toxic release. Air cylinders brought from off-site locations may be used to extend capacity beyond six hours.

2.1.6 Stored Gases

Table 2.1-14 lists the vessels used for storage of pressurized gas at BVPS-1. The service operating, design and maximum pressure, location of vessel, and total energy stored are shown in the table.

All storage vessels, except for propane gas storage and air storage tanks for the diesel generator, are not located adjacent to equipment essential for maintaining a safe reactor shutdown.

Nitrogen makeup is provided by a tank truck supply located adjacent to the South Coolant Recovery Tank Cubicle (BR-TK-4B).

Missiles generated by the propane storage tanks and the air storage tanks in the diesel generator structure are discussed in Section 5.2.6.

All storage vessels have provisions for relief protection. This protection precludes any missiles generated from accidental rupture of tanks caused by overpressurization.

The vessels are protected from truck lanes or heavy vehicle traffic. No heavy loads are transported over vessel storage areas.

There are no exceptions or deviations taken to Occupational Health Administration OSHA 29 CFR 1910 Subpart H-Hazardous Material Sections 1910.101 Compressed Gases, 1910.103 Hydrogen and 1910.104 Oxygen, Subpart M - Compressed Gas and Compressed

constant static air pressure in the ductwork downstream of VS-F-17. This constant static air pressure will therefore maintain constant static pressures in the normal switchgear and rod control M/G rooms to help maintain the Control Room process rack pressure boundary criteria.

In the event of a loss of power under either normal operating or accident conditions, all electrically powered motors and controls associated with the air conditioning and pressurizing systems are furnished with emergency power from the emergency diesel generators.

Refer to Section 8.5.3 for a discussion of the emergency battery room ventilation system and smoke detection system.

In the event of a fire in any of the individual zones in the main control area, separate zone smoke dampers are manually actuated to seal off the affected zone and permit the remaining zones to continue to function. Provision is also made for purging smoke from the fire area by positioning the proper outside air and exhaust air dampers. See Section 9.10.2 for a discussion of the control room ventilation systems smoke detectors and temporary ventilation provisions.

Section 2.1.5 discusses the storage of chlorine at BVPS-1. Released chlorine is toxic to personnel. To continue safe operation or to proceed through and maintain a safe shutdown, the control room ventilation must be free of high concentrations of chlorine. This is achieved by the remoteness of the control room from the chlorine storage area, the separation provided by the turbine building's greater elevation than the control room, and the control room's controlled restricted air intake. Considering the worst accident and meteorological conditions, the control room pressurization system could be actuated to prevent infiltration of chlorine.

Three seismically qualified chlorine detectors sample the control room air intake. These detectors will alarm and automatically initiate control room isolation and emergency bottled air supply upon sensing a chlorine concentration in excess of a setpoint of 5 ppm.

Refer to Section 11.3.5 for a discussion of the Control Room Area Radiation Monitors (RM-RM-218A,B).

Following the first hour, self-contained breathing apparatus units and sufficient reserve air cylinders are available to support the minimum control room shift composition for at least five additional hours. This satisfies Regulatory Guides 1.78 and 1.95. Sufficient additional units are provided to support the members of the emergency squad stationed outside the control room for one hour, after which these personnel would move away from the area affected by the toxic release. Air cylinders brought from off-site locations may be used to extend capacity beyond six hours.

The main control area is thus designed to be continuously habitable under any foreseeable condition of operation.

ATTACHMENT 5

DBNPS USAR Changes

Relocation of Regulatory Guide Information for Davis-Besse

1. Regulatory Guide 1.29 (Revision 2, February 1976) [Insert "A" in USAR Section 3.2.1.1]

   Seismic classification complies with Regulatory Guide 1.29 (Revision 2, February 1976) as described below.

2. Regulatory Guide 1.26 (Revision 3, February 1976) [Insert "B" in USAR Section 3.2.2]

   Quality group classification complies with Regulatory Guide 1.26 (Revision 3, February 1976) as described below.

3. Regulatory Guide 1.54 (Revision 0, June 1973) [Insert "C" in USAR Section 3.8.2.1.11]

   1. Davis-Besse commits to the regulatory position of this Guide with the following clarifications:

   a. This Regulatory Guide and its associated ANSI Standard implies that a significant amount of coating work is required at the plant site. Although this is correct for construction sites, the coating work at an operating site generally consists of repair and touchup work following maintenance and repair activities or the initial coating of components such as hangers, supports, and piping during facility modifications. Therefore, in lieu of the full requirements of this Regulatory Guide and ANSI N101.4, Davis-Besse shall impose the following requirements:

   1) The quality assurance requirements of Section 3 of ANSI N101.4 applicable to the coating manufacturer shall be imposed on the coating manufacturer through the procurement process.

   2) Coating application procedures shall be developed based on the manufacturer's recommendations for application of the selected coating systems.

   3) Coating applicators shall be qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations.

   4) Quality control personnel shall perform inspections to verify conformance of the coating application procedures. Section 6 of ANSI N101.4 shall be used as guidelines in the establishment of the inspection program.

   5) Quality control personnel shall be qualified to the requirements of Regulatory Guide 1.58 (Revision 1).

   6) Documentation demonstrating conformance to the above requirements shall be maintained.

Relocation of Regulatory Guide Information for Davis-Besse

b. *The requirements of Position A of this Guide apply to surfaces within containment with the following exceptions:*

   1) *Surfaces to be insulated.*

   2) *Surfaces contained within a cabinet or enclosure.*

   3) *Repair/touchup areas less than 30 square inches or surface areas such as: cut ends; bolt heads, nuts and miscellaneous fasteners; and damage resulting from spot, tack or arc welding.*

   4) *Small items such as small motors, handwheels, electrical cabinets, control panels, loud speakers, motor operators, etc. where special painting requirements would be impracticable.*

   5) *Stainless steel or galvanized surfaces.*

   6) *Banding used for insulated pipe.*

2. *Davis-Besse commits to the requirements of ANSI N101.4-1972 for activities comparable in nature and extent to construction phase activities as modified by the commitment to Regulatory Guide 1.54.*

4. Regulatory Guide 1.55 (Revision 1, September 1980)

   This Regulatory Guide is only applicable to the Perry Nuclear Power Plant and will therefore not be relocated to the Davis-Besse USAR.

5. Regulatory Guide 1.78 (June 1974) [Insert "D" in USAR Section 6.4.2]

   Davis-Besse commits to the regulatory position of Regulatory Guide 1.78 (June 1974).

6. Regulatory Guide 4.15 (Revision 1, February 1979) [Insert "E" in USAR Sections 11.2, 11.3, 11.4, 11.5, 11.6 and 12.4]

   Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979).

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

The structures, components, and systems which are required to avoid or mitigate the consequences of abnormal operational transients or accidents are classified in this section. | 20

3.2.1 Seismic Classification

The USAR terminology used for defining the earthquakes and the seismic classification of the equipment is the same as was used in the PSAR and FSAR. This has been done to provide continuity between the various rationale, criteria, and design commitments made in the PSAR, FSAR, and the final design evaluations presented in the USAR.

The Maximum Probable Earthquake is 0.08g. It is the conservatively determined earthquake and associated ground motion that might reasonably or probably be expected to occur at the nuclear plant site. The Maximum Probable Earthquake is similar to the Operating Basis Earthquake (OBE) terminology presently being used by the NRC. | 21

The Maximum Possible Earthquake is 0.15g. It is the conservatively determined earthquake and associated ground motion which could conceivably or possibly occur at the site. The Maximum Possible Earthquake is similar to the Safe Shutdown Earthquake (SSE) terminology presently being used by the NRC. | 4

3.2.1.1 Definitions A

Class I structures, systems, and components for seismic design purposes are defined (in General Design Criterion 2 of Appendix A to 10CFR50, Appendix A to 10CFR100, and NRC Reg. Guide 1.29 [Rev. 2, 2/76]) as those structures, systems, and components important to safety that are designed to remain functional in the event of a Maximum Possible Earthquake. These structures, systems, and components are those necessary to ensure:

a. The integrity of the reactor coolant pressure boundary.

b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or

c. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

Class I structures, systems and components are also defined as nuclear safety related (Q) and are relied upon to remain functional during design basis events. | 22

Class II structures, systems, and components are defined as those structures, systems, and components which are not classified as Class I. These structures, systems, and components are designed in accordance with the Uniform Building Code, 1967 edition for Seismic Zone 1 loads. Load combinations and allowable stresses for Class II structures are described in Subsections 3.8.1.3.2 and 3.8.1.5, respectively.

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3.2.1.3 Partially Class I Structures and Systems

All Class I structures are separated by an expansion joint from all Class II structures. Class I equipment and systems located in Class II structures have reinforced concrete enclosures designed to withstand the loads for Class I structures. Accordingly these Class II structures are designated as partially Class I structures.

3.2.2 System Quality Group Classification B

Table 3.2-2 delineates the system Quality Group classifications of each component of those fluid systems that are required to prevent or mitigate the consequences of accidents or malfunctions within a reactor coolant pressure boundary or to permit safe shutdown of the reactor and maintenance with safe shutdown condition. The containment structure pressure boundary is shown in Figure 3.2-1.

The system piping and instrument diagrams for the fluid systems delineate, with the symbol "Q", the boundary of all "Q" listed components. "Q" listed components are as defined in subparagraph 3.2.1.1. 18 22

The extent and configuration of overpressure protection provided for systems and components relative to referenced codes and standards are illustrated on the piping and instrumentation diagrams. 18

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The allowable local stresses in the Containment Vessel are within those allowed by the ASME Boiler and Pressure Vessel Code for Class B Nuclear Vessels.

In the event that any pipe does not meet the above criteria, guard pipes are provided to direct the effects of the pipe rupture into the Containment Vessel.

3.8.2.1.11 Containment Vessel Painting C

Two coating systems are currently used within the containment.

The first system consists of epoxy or modified phenolic coatings, such as Amercoat No. 66 manufactured by the Amercoat Corporation, or Phenoline No. 305 finish manufactured by the Carboline Company, or Val-Chem Hi-Build Epoxy manufactured by the Mobile Chemical Company, or approved equal coating. This system is applied to concrete floors, walls, and ceilings.

The second system is an inorganic zinc primer followed by an organic topcoat, such as Dimetcote No. 4 or No. 6 primer followed by Amercoat No. 66 epoxy topcoat as manufactured by Amercoat Corporation, or Carbo-Zinc No. 11 primer followed by Phenoline No. 305 modified phenolic finish, as manufactured by the Carboline Company, or Mobilzinc 7 primer followed by Val-Chem Hi-Build epoxy, as manufactured by Mobil Chemical Company, or approved equal systems. This system is applied to ferrous metal surfaces, such as structural steel, liner plate, piping, and equipment within the entire containment boundary and up to wainscot height above the floor levels in areas subject to hard usage or to contamination.

The function of the materials is to provide surfaces which resist exposures due to both normal operating and LOCA conditions. Exposures include ionizing radiation, high temperature, and impingement from sprays.

Physical characteristics of the materials are as follows:

1. Application characteristics: Pot life, drying and recoating times, all at 70°F, and contents of solids are as follows:

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6.4 HABITABILITY SYSTEMS

21

Control room systems are designed so that habitability of the control room can be maintained under normal and accident conditions in accordance with the general guidance in General Design Criteria 19 of 10CFR 50, Appendix A. The control room ventilation systems are described in Section 9.4.1.

6.4.1 Radiation Monitoring

The radiation shielding and control room layout are described in Section 12.1. Control room airborne radioactivity monitoring is described in Section 12.2. The evaluation of radiological exposure to control room personnel for postulated accident conditions are presented in Section 15.4.

6.4.2 Toxic Gas Protection Provisions

D

The habitability of the control room was evaluated using procedures described in Regulatory Guide 1.78, "Assumptions for Evaluation of the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." As indicated in Section 2.2, analysis of off-site storage or transport of chemicals and hazardous materials stored onsite demonstrate that no toxic or explosive materials are stored in volumes or locations which pose a control room habitability hazard exceeding emergency system capabilities. Administrative procedures are in place to control the allowable amount of transient hazardous materials in the vicinity of the control room. A sodium hypochlorite biocide system is used, thus eliminating an onsite chlorine hazard, therefore, special protection against toxic gases will not be required. Self-contained breathing apparatus is provided for the emergency crew to provide assurance of control room habitability in the event of occurrences such as smoke hazards.

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11.2 LIQUID WASTE SYSTEMS

11.2.1 Design Objectives

The systems handling liquid wastes are designed such that the estimated releases in liquid effluents comply with the following requirements of 10CFR20 and 10CFR50:

a. The individual radionuclide concentrations in liquid effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10CFR20.

b. The releases of radioactivity from the station shall comply with the "as low as reasonably achievable" standard set forth in 10CFR50.

11.2.2 System Descriptions

E

11.2.2.1 Process Arrangement

11.2.2.1.1 Clean Liquid Radwaste System

The functional drawing for this system is shown in Figure 11.2-2. A list of major components, as well as design information for each, is contained in Table 11.2-1. Figure 11.2-4 is a schematic of the process cycle. 14

11.2.2.1.2 Miscellaneous Liquid Radwaste System

The functional drawing for this system is shown in Figure 11.2-3. A list of major components, as well as design information for each, is contained in Table 11.2-2. 14

18

11.2.2.2 Waste Sources

Radiation sources, activities, quantities and concentrations are dependent upon various factors such as, the fuel load (enrichment of fuel) and the operating cycle length. The current information is maintained in the ODCM and the source term analysis for the current operating cycle reload report. 20

11.2.2.2.1 Clean Liquid Radwaste System

The major source of waste for this system is reactor coolant letdown resulting from boron dilution operations or from coolant expansion during reactor startups. Other sources include leakage, drainage, and relief flows from valves and equipment containing reactor-grade liquid. 20

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11.3 GASEOUS WASTE SYSTEM

11.3.1 Design Objectives

The system is designed so that estimated releases of gaseous effluents from the station comply with the following requirements of 10CFR20 and 10CFR50:

a. The individual radionuclide concentrations in gaseous effluents at the site boundary shall not exceed the limits from releases to unrestricted areas given in Appendix B of 10CFR20.

b. The releases of radioactivity from the station shall comply with the as low as reasonably achievable standard set forth in 10CFR50.

11.3.2 System Description

E

11.3.2.1 Process Arrangement

The functional drawing for the portion of this system handling unaerated hydrogenated gases and some of the cover gases is shown in Figure 11.3-1. A list of major components, as well as design information for each, is contained in Table 11.3-1. A flow schematic of the entire system is shown in Figure 11.3-2. 14

11.3.2.2 Waste Sources

Radiation sources, activities, quantities and concentrations are dependent upon various factors such as, the fuel load (enrichment of fuel) and the operating cycle length. The current information is maintained in the ODCM and the source term analysis for the current cycle reload report. 20

11.3.2.2.1 Unaerated Hydrogen-Containing Gas

These gases are the primary concern of this system. They are present in relatively large quantities and contain most of the gaseous radioactivity that will be processed. In general, they contain minimal oxygen and often large amounts of hydrogen.

Sources include off-gas from the Reactor Coolant System in the Clean Waste Receiver Tanks, the gases vented from the boric acid evaporators (if aligned to the gaseous radwaste system) and reactor coolant drain tank. Significant quantities of waste can also be produced by such infrequent events as the venting of the makeup tank or pressurizer or the relieving of the quench tank relief valve. None of the components is specifically designed to withstand a hydrogen explosion for the following reasons: 19 21 20

a. The condition under which a hydrogen explosion may take place will never arise. That condition is greater than 5 percent oxygen by volume in a hydrogen-rich atmosphere.

b. There is no source input to the waste gas surge tank which contains sufficient amounts of oxygen. The major waste sources originate in primary reactor coolant which has been calculated to contain much less than 1 percent oxygen (see Subsection 9.3.2).

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 Design Objectives (E)

The radiological monitoring system was designed to:

a. Continuously detect and record the level of radioactivity in certain process streams and all station effluent streams to ensure compliance with the requirements of 10CFR20, 10CFR50 and Safety Guide Number 21.

b. Provide operating personnel with a continuous indication of the beta and gamma-emitting radioactivity in selected station areas during both normal operation and postulated occurrences. |16

c. Provide alarm indication on all monitored points for activity level increases to a calculated maximum tolerance level and initiate protective functions, as required, on activity level increases above these tolerance values. All activity level measurements are acted upon; no discrimination is made between normal operational conditions and accident situations.

The radiological monitoring system provided to fulfill the above design objectives is outlined in Table 11.4-1.

Table 11.4-1 lists the process and effluent radioactivity monitors, specifying the detector types, measurement ranges, and sensitivities. Detector sensitivities are stated with respect to the predominant radionuclide in the system to be monitored. Liquid monitors are described first, then gaseous and airborne monitors. |16

11.4.2 Continuous Monitoring

11.4.2.1 General Design

Each channel of the process radioactivity monitoring system consists of remotely-located, interconnected subsystems. The detectors are located within the station adjacent to the monitored process. The control, readout, recording, and power supply instrumentation is installed in the control cabinet room with alarm indication located in the main control room. Each monitor channel has, along with the indicators designated in Subsection 11.4.2.2, an indicator in the cabinet room. Each monitor channel has alarm functions along with complementary switches to initiate operational functions on high radioactivity levels. |7

The system is redundant through the measurement of the station effluents, both at their point of release and at their potential radioactivity source points within the station. A duplication of instruments, to increase the level of confidence of measurements, is provided for certain station process and effluent streams.

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11.5 SOLID WASTE SYSTEM

11.5.1 Design Objectives

E

The solid waste system was designed to receive, process, package, and store all of the solid wastes generated at the Davis-Besse Nuclear Power Station. The system was designed to accomplish this task in a manner so simple as to minimize the possibility of radioactive material release and personnel exposure.

The materials handled by the solid waste system include bead-type resins, spent filter cartridges, powdered resins, and miscellaneous solid waste such as paper, rags, contaminated clothing, gloves and shoe coverings.

The solid waste system area was designed to provide the necessary shielding to prevent the overexposure of operating personnel to radioactive sources. This is accomplished through the use of lead shielding, concrete shielding, and safe operating procedures. 19

11.5.2 System Inputs

This section is the original design basis for the solid radwaste system, and as such, has been left as originally written.

As previously noted, the primary sources of solid radwaste are:

Spent bead resins
Spent powdered resins 9
Spent filter cartridges
Misc. paper, clothing, etc.

Table 11.5-1 gives estimates of the total maximum and average quantities of the various radionuclides, excluding tritium contributed annually by each of 7
these sources during a fuel cycle. In compiling this table, it was generally assumed that the estimates of maximum and average radionuclide inventories were based, respectively, on 1% and 0.1% failed fuel conditions occurring 7
annually during an equilibrium fuel cycle. Whenever reactor coolant specific activities were needed for calculation, the values used for dissolved radionuclides were the maximum levels, corrected to correspond to a coolant density of 1 g/ml, attained annually in the primary system during an 7
equilibrium fuel cycle. For corrosion products, the values given in Table 11.1-7, also modified to correspond to a liquid density of 1 g/ml, were used. The estimated quantities (volumes) of the above wastes are given in Table 11.5-2.

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11.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

11.6.1 Background Radiation

E

The preoperational radiological monitoring program commenced on August, 1972 and provided approximately five full years of background data prior to station operation.

Concentrations of background radioactivity for the Davis-Besse site are listed in Table 11.6-1. These concentrations are based on data from the 1981 annual radiological environmental monitoring program. | 14

The major objective of the preoperational monitoring program was to accumulate data for two years on the background radiation at the Davis-Besse site. In an effort to make the program as comprehensive and up-to-date as possible, minor changes were made as necessary to incorporate proposed future monitoring requirements. The program complied with Regulatory Guide 4.8, "Environmental Technical Specification For Nuclear Power Plants." | 14

The operational radiological monitoring program is similar to the pre-operational program. | 7

11.6.2 Critical Pathways

The design bases for the critical pathways for exposure to humans due to the small quantity of radioactivity released from the station during operation are described in Appendix 11A together with the calculational models used in arriving at exposure estimates to the populace. | 20

11.6.3 Sampling Media, Locations and Frequency

Ambient radiation measurements are made using thermoluminescent dosimeters (TLD). Dosimeter packets are located at each of twenty-seven locations. Each dosimeter consists of up to four individual exposure areas to provide a statistical analysis of the data. The dosimeter packets at each location are changed quarterly. There are five locations for determining airborne particulates and iodine which are changed weekly. Iodine is one of the most restrictive radionuclides which could be released to the environment. Weekly surface water samples are collected from two locations. These samples should provide the first indication of radioactive effluent leakage into the environment from the station discharge. Drinking water samples are taken from two public water supplies to detect any increases over background radioactivity concentrations. Ground water samples are collected quarterly from one location. Two species of fish are sampled annually at two locations because they are a food source and certain radionuclides concentrate in their flesh. Broad leaf vegetation samples are collected monthly, when available, since this is a direct pathway to man, should any radionuclides be present in the crops. Milk samples are collected semi-monthly during the grazing season (May-Oct.) and monthly at other times (Nov.-Apr.). Samples are collected from the milking animals closest to the station and from milking animals at least 15-30 km away. Milk is analyzed since radioactive iodine in air can be concentrated by the milk exposure pathway. | 14 | 21 | 18

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12.4 RADIOACTIVE MATERIALS SAFETY

E

12.4.1 Materials Safety Program

Fuel storage and handling is described in Section 9.1. The Radwaste Management Program is described in Chapter 11. This section describes the Materials Safety Program for the remainder of the radioactive materials to be used at the site.

All licensed source and byproduct materials used for calibration and sample analysis are kept in a shielded container when not in use. These containers are posted or labeled in accordance with 10CFR20 regulations and are kept locked when not in use to prevent removal of sources by unauthorized persons. When not in use, the sources in their containers are stored in a locked designated storage area. 7 14

Any Non-Destructive Test (NDT) sources which are brought on site are controlled by the contractor who will do the testing. The contractor is then required to comply with those regulations which are applicable to him to assure adequate materials safety.

12.4.2 Facilities and Equipment

The Hot Laboratory is designed for handling radioactive materials and is maintained at a negative pressure to prevent airborne particles from leaving the area. The fume hoods in the Hot Laboratory, as well as the ventilation for the room, vent through a prefilter/absolute-filter/charcoal filtering system. There is an area monitor located in the Hot Laboratory, and its output is continuously recorded. Equipment, glassware, and tools used in working with radioactive materials are suitably controlled and are not used in clean areas unless the item has been surveyed and meets the limits for uncontrolled areas. 7 14

Sealed neutron and gamma sources used for calibration are stored in designated storage locations controlled by Radiation Protection personnel. Portable instrumentation is calibrated in the Instrument Calibration Room or other designated low background radiation areas. Radiation sources used at the location of permanently installed monitoring equipment, such as area radiation monitors and reactor instrumentation, will be manipulated with remote handling devices and shields to maintain personnel exposures as low as reasonably achievable, when necessary. 7 16 7 14

Remote handling devices and shields are available to be used when handling radiation sources. Licensed gamma sources are stored in lead or steel containers. Licensed neutron sources are contained in a hydrogenous shield when being transported or stored. 7 19

12.4.3 Personnel and Procedures

Section 13.2 describes the training program implemented to ensure that only well-trained personnel handle licensed radioactive sources and byproduct material. These persons have the proper personnel dosimetry and survey the source upon removal from its container. Sealed sources are leak-tested semi-

ATTACHMENT 6

PNPP USAR Changes

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	Reference
1.24 - (Revision 0 - 3/72;RRRC Cat. 1)		
Assumptions used for evaluating the potential radiological consequences of a pressurized water reactor gas storage tank failure	Not applicable to PNPP design.	-
1.25 - (Revision 0 - 3/72;RRRC Cat. 1)		
Assumptions used for evaluating the potential radiological consequences of a fuel handling accident in the fuel handling and storage facility for boiling and pressurized water reactors	PNPP design conforms to this guide with the following exceptions: a. (Regulatory Position C.1.j) filter efficiencies of 95% are used in accordance with Regulatory Guide 1.52; b. (Regulatory Position C.3.a/c) dose conversion factors and average gamma energies are taken from NRC TACT III and/or TACT 5 computer code in lieu of Table 1 and Reference 12.	6.5.1, 9.1.2, 9.4.2, 15.7.4
1.26 - (Revision 3 - 2/76;RRRC Cat. 1)		
Quality group classifications and standards for water-, steam- and radioactive-waste-containing components of nuclear power plants	~~See Table 1.8-2.~~ PNPP design complies with this guide.	3.2.1, Table 3.2-1, 6.2.4, 6.5, 6.7, 9.4, 9.5, 10.3.3, ~~3.7.2~~

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.:RRRC Category)	Degree of Conformance	Reference
1.27 - (Revision 2 - 1/76:RRRC Cat. 2)		
Ultimate heat sink for nuclear power plants	PNPP conforms with this guide with the following clarification: Technical Specifications do not address the loss of capability of the ultimate heat sink since there is no single active or passive failure which would preclude the ultimate heat sink from meeting its design criteria.	2.4, 9.2.5
1.28 - (Revision 2 - 2/79)		
Quality assurance requirements (design and construction)	See Table 1.8-2.	17.2
1.29 - (Revision 3 - 9/78:RRRC Cat. 1)		
Seismic design classification	~~See Table 1.8-2.~~ Insert text from USAR Pages 1.8-65a through 1.8-65c.	3.2.1, Table 3.2-1, 3.7.3, 6.2.4, 6.5, 6.7, 8.3.1, 9.1, 9.3.5, 9.4, 9.5, 10.3.1,

TABLE 1.8-2 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	Reference
1.26 - (Revision 3 - R - 3/76;RRRC Cat. 1)		
Quality group classifications and standards for water-, steam- and radioactive-waste-containing components of nuclear power plants	PNPP design complies with this guide.	3.2.1, Table 3.2-1, 6.2.4, 6.5, 6.7, 9.4, 9.5, 10.3.3, 17.2
1.28 - (Revision 2 - 2/79)		
Quality assurance requirements (design and construction)	PNPP complies with this guide.	17.2
~~1.29 - (revision 3 - 9/78;RRRC Cat. 1)~~		
~~Seismic design classification~~	PNPP design complies with this guide, with exceptions as stated in Notes 19 and 24 of Table 3.2-1 and with the following clarifications: Position C.1.e - The design of the main steam system incorporates a third isolation valve between the outermost MSIV and the turbine stop valve in each main steam line. The piping downstream of this MOV is nonsafety class	3.2.1, Table 3.2-1, 3.7.3, 6.2.4, 6.5, 6.7, 8.3.1, 9.1, 9.3.5, 9.4, 9.5, 10.3.1, ~~17.2~~

Not this change

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TABLE 1.8-2 (Continued)

Regulatory Guide (Rev.:RRRC Category)	Degree of Conformance	Reference
1.29 (Continued)	Position C.3 and C.4 - Seismic Category I design requirements are required to be extended "to the first seismic restraint beyond the defined boundaries." Seismic analysis of a piping system requires division of the system into discrete segments terminated by fixed points. Thus the seismic restraint is not terminated at a seismic restraint, but is extended to the first point in the system, that can be treated as an anchor to the plant structure or to a distance sufficient such that the effects of the piping beyond the safety class boundary are insignificant. Paragraph C.4 also requires that "the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 be applied to the safety requirements" of such items. Both these requirements are considered to be adequately met by the following practice: a. Design and design control for these items are carried out in the same manner as that for items directly important to safety. This includes the performance of appropriate design reviews.	

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TABLE 1.8-2 (Continued)

Regulatory Guide (Rev.:RRRC Category)	Degree of Conformance	Reference
1.29 (Continued)	Position C-4 - Design for items that would otherwise be classified as non-seismic but whose failure could reduce the functioning of items important to safety to an unacceptable safety level is performed in accordance with Seismic Category I requirements. Design control is carried out in the same manner as that for items directly important to safety. For piping and support of piping beyond the class break the following applies: a. Procurement of piping, inline components and their supports is performed in accordance with the item's safety classification, i.e., nonsafety. b. Installation of piping and inline components is also performed as with other nonsafety items. c. Final installation of component supports is inspected as a formal part of the Corporate Nuclear Quality Assurance Program.	

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TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.:RRRC Category)	Degree of Conformance	Reference
1.52 - (Revision 2 - 3/78;RRRC Cat. 2)		
Design, testing and maintenance criteria for postaccident engineered-safety-feature atmosphere cleanup system air filtration and absorption units of light-water-cooled nuclear power plants	PNPP design and testing conform to this guide as presented in Tables 6.5-1 through 6.5-3.	6.4, 6.5.1, 9.1, 9.4, 12.3, 15.7, Tech. Specs.
1.53 - (Revision 0 - 6/73;RRRC Cat. 1)		
Application of single failure criterion to nuclear power plant protection systems	Single failure criteria is applied to protection systems in accordance with Regulatory Guide 1.53.	6.5.3, 7.2.2, 7.3.2, 7.4.2, 7.6.2, 8.1, 9.4
1.54 - (Revision 0 - 6/73;RRRC Cat. 1)		
Quality Assurance requirements for protective coatings applied to water-cooled nuclear power plants	~~See Table 1.8-2.~~ INSERT B	6.1.1, 6.1.2, ~~17.2~~ e
1.55 - (Revision 0 - 6/73;RRRC Cat. 1)		
Concrete placement in Category I structures	~~See Table 1.8-2.~~ PNPP commits to the regulatory position of this guide for activities that are comparable in nature to construction phase activities.	3.8, ~~17.2~~ e

INSERT B

PNPP commits to the regulatory position of this Guide with the following clarifications:

a. This Regulatory Guide and its associated ANSI Standard implies that a significant amount of coating work is required at the plant site. Although this is correct for construction sites, the coating work at an operating site generally consists of repair and touchup work following maintenance and repair activities or the initial coating of components such as hangers, supports, and piping during facility modifications. Therefore, in lieu of the full requirements of this Regulatory Guide and ANSI N101.4, PNPP imposes the following requirements:

1) The quality assurance requirements of Section 3 of ANSI N101.4 applicable to the coating manufacturer shall be imposed on the coating manufacturer through the procurement process.

2) Coating application procedures shall be developed based on the manufacturer's recommendations for application of the selected coating systems.

3) Coating applicators shall be qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations.

4) Quality control personnel shall perform inspections to verify conformance of the coating application procedures. Section 6 of ANSI N101.4 shall be used as guidelines in the establishment of the inspection program.

5) Quality control personnel shall be qualified to the requirements of Regulatory Guide 1.58 (Revision 1).

6) Documentation demonstrating conformance to the above requirements shall be maintained.

b. The requirements of Position A of this Guide apply to surfaces within containment with the following exceptions:

1) Surfaces to be insulated.

2) Surfaces contained within a cabinet or enclosure.

3) Repair/touchup areas less than 30 square inches or surface areas such as: cut ends; bolt heads, nuts and miscellaneous fasteners; and damage resulting from spot, tack or arc welding.

4) Small items such as small motors, handwheels, electrical cabinets, control panels, loud speakers, motor operators, etc. where special painting requirements would be impracticable.

5) Stainless steel or galvanized surfaces.

6) Banding used for insulated pipe.

PNPP commits to the requirements of ANSI N101.4-1972 for activities comparable in nature and extent to construction phase activities.

TABLE 1.8-1 (Continued)

Regulatory Guide (Rev.;RRRC Category)	Degree of Conformance	Reference
1.75 - (Revision 2 - 9/78;RRRC Cat. 4)		
Physical independence of electrical systems	PNPP design is in accordance with IEEE Standard 384-1974, as modified by the positions of Regulatory Guide 1.75, with the alternative positions as discussed in Table 8.1-2.	7.1.2, 7.6.1, 8.1, 8.3.1
1.76 - (Revision 0 - 4/74;RRRC Cat. 4)		
Design basis tornado for nuclear power plants	PNPP design conforms to this guide.	2.3.1, Table 2.3-5, 3.3.2 3.5.1.4
1.77 - (Revision 0 - 5/74;RRRC Cat. 1)		
Assumptions used for evaluating a control rod ejection accident for pressurized water reactors	Not applicable to the PNPP design.	-
1.78 - (Revision 0 - 6/74;RRRC Cat. 1)		
Assumptions for evaluating the habitability of a nuclear power plant control room during a postulated hazardous chemical release	PNPP design conforms to this guide.	2.2.3, 6.4 For information only

ENCLOSURE

DBNPS Commitment List

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THIS SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER-REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENT	DUE DATE
None	**N/A**