UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 7 1984

Docket No. 50-354

APPLICANT: Public Service Electric & Gas Company (PSE&G)

FACILITY: Hope Creek Generating Station

SUBJECT: SUMMARY OF QUALITY ASSURANCE (QA) MEETING

On July 18, 1984, a meeting was held in the Bethesda, Maryland offices of the NRC to discuss the Hope Creek QA Program. A list of attendees is included as Enclosure 1 to this meeting summary.

The basis of the meeting was a letter dated June 12, 1984, (A. Schwencer, NRC to R.L. Mittl, PSE&G) which transmitted a list of outstanding QA issues (through FSAR Amentment 5) to PSE&G. Additional issues were added as a result of the review of FSAR Amendment 6 which was submitted on June 28, 1984. Enclosure 2 is the (staff-generated) agenda used at the meeting.

Enclosure 3 is the information provided by PSE&G at the meeting. Based on the meeting discussions PSE&G responses to the outstanding issues are acceptable, except for the following:

Item 1(b)

PSE&G will clarify the commitment to Reg. Guide 1.64.

Item 1(c)

PSE&G will clarify the commitment to Reg. Guide 1.144.

Item 6

PSE&G will insert commitment into FSAR addressing the shipping and packaging of code items per Reg. Guide 1.38.

Item 11(a)(1)

PSE&G will add to footnote to Table 3.2-1 stating that reactor pressure vessel internal structures which are accessible are included in the ISI Program which is covered by the operational QA program.

Item 11(a)(2)

Regarding the spent fuel pool liner, PSE&G will note in the FSAR that repairs to the liner will be included in the operational QA program.

Item 11(a)(4)

PSE&G will state in FSAR that the operational QA program is applicable to these valves.

Item 11(a)(5)

PSE&G will add footnote noting that functionally will be verified by routine surveillance by the QA organization.

Item 11(b)

PSE&G will revise table to indicate that sheetpile ball is included and that Hope Creek no longer has quarrystone revetments.

Item 11(c)

PSE&G will include the valves in the table.

Item 11(e)

PSE&G will resubmit response and commit to operational QA program coverage.

Item 12

PSE&G will delete the exceptions noted in FSAR Table 1.11-1

PSE&G representatives indicated that the responses discussed at the meeting will be formally submitted to the NRC by August 1, 1984, and the indicated FSAR changes would be incorporated in FSAR Amendment 8.

David H. Wagner, Project Manager
Licensing Branch No. 2
Division of Licensing

Enclosures: As stated

cc: See next page

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DL:LB#2/PM	DL:LB#2/BC
DWagner:bdm	ASchwencer
8/7/84	8/7/84

Distribution:
Docket File
PRC System
LB#2 Reading
D. Wagner
P. Shuttleworth
J. Spraul
L. Dewey, OELD
Local PDR
NSIC
NRC PDR
A. Schwencer

Hope Creek

Mr. R. L. Mittl, General Manager
Nuclear Assurance & Regulation
Public Service Electric & Gas Company
80 Park Plaza T22A
Newark, New Jersey 07101

cc:
Troy B. Conner, Jr. Esquire
Conner & Wetterhahn
1747 Pennsylvania Aveneu N.W.
Washington, D.C. 20006

Richard Fryling, Jr., Esquire
Associate General Solicitor
Public Service Electric & Gas Co.
80 Park Plaza T5E
Newark, New Jersey 07101

Mr. P.R.H. Landrieu
Project Manager - Hope Creek
Public Service Electric & Gas Co.
80 Park Plaza T17A
Newark, New Jersey 07101

Richard F. Engel
Deputy Attorney General
Division of Law
Environmental Protection Section
Richard J. Hughes Justice Complex CN-112
Trenton, New Jersey 08625

Mr. David A. Caccia
Box 70, A.R.D. #2
Sewell, New Jersey 08080

Mr. R. P. Douglas
Manager-Licensing & Analysis
Public Service Electric & Gas Co.
80 Park Plaza T22A
Newark, New Jersey 07101

Mr. B. G. Markowitz, Project Manager
Bechtel Power Corporation
50 Beale Street
P. O. Box 3965
San Francisco, California 94119

Susan C. Remis
Division of Public Interest Advocacy
New Jersey State Department of
the Public Advocate
Richard J. Hughes Justice Complex
CN-850
Trenton, New Jersey 08625

Office of Legal Counsel
Department of Natural Resources
and Environmental Control
89 Kings Highway
P.O. Box 1401
Dover, Delaware 19903

Mr. K. W. Burrowes, Project Engineer
Bechtel Power Corporation
50 Beale Street
P. O. Box 3965
San Francisco, California 94119

Mr. W. H. Bateman
Resident Inspector
U.S.N.R.C.
P. O. Box 241
Hancocks Bridge, New Jersey 08038

Mr. J. M. Ashley
Senior Licensing Engineer
c/o PSE&G Company
Bethesda Office Center, Suite 550
4520 East-West Highway
Bethesda, Maryland 20814

Office of the Director
DNREC - Division of Environmental Control
89 Kings Highway
P. O. Box 1401
Dover, Delaware 19903

Mr. R. S. Salvesen
General Manager-Hope Creek Operation
Public Service Electric & Gas Co.
P.O. Box A
Hancocks Bridge, New Jersey 08038

Mr. B. A. Preston
Principal Engineer
Public Service Electric & Gas Co.
80 Park Plaza T22A
Newark, New Jersey 07101

Mr. A.E. Giardino
Manager - Quality Assurance E&C
Public Service Electric & Gas Co.
P.O. Box A
Hancocks Bridge, New Jersey 08038

State House Annex
ATTN: Deputy Attorney General
State of New Jersey
36 West State Street
Trenton, New Jersey 08625

Commissioner
Department of Public Utilities
State of New Jersey
101 Commerce Street
Newark, New Jersey 07102

Ms. Mary Henderson, Clerk
Lower Alloways Creek Township
Salem County, New Jersey 08079

U.S. Environmental Protection Agency
ATTN: EIS Coordinator
Region II
26 Federal Place
New York, New York 10007

Enclosure 1

MEETING TITLE: Quality Assurance Meeting

APPLICANT: Public Service Electric & Gas Company

FACILITY: Hope Creek Generating Station

DATE: Wednesday, July 18, 1984

NAME	AFFILIATION
Frank Omohundro	PSE&G
W. R. Schultz	PSE&G
Paul A. Benini	PSE&G
Bruce Preston	PSE&G
William Gailey	PSE&G
C. P. Johnson	PSE&G
Dave Wagner	NRC
Jack Spraul	NRC

LICENSEE-NRC MEETING AGENDA
JULY 18, 1984
HOPE CREEK GENERATING STATION
OUTSTANDING QA ISSUES THRU FSAR AMEND. 6

1. Commitments to Regulatory Guides in Section 1.8 need revision as noted below:

   a. Commitment to Regulatory Guide 1.33 needs clarification regarding "event based" vs "functional" emergency procedures.

   b. Commitment to Regulatory Guides 1.39 and 1.64 needs to address the operations phase.

   c. Commitment to Regulatory Guide 1.94 and 1.144 needs to be clarified to show commitment during the operations phase.

   d. First paragraph of the commitment to Regulatory Guide 1.123 needs grammatical clarification. Item a clarification unacceptable during the operations phase. Delete or provide additional justification.

2. The first sentence of the third paragraph of Section 17.2.16 needs clarification.

3. "As applicable" on the third line of page 17.2.18 needs to be deleted or defined.

4. "Periodically" in the last paragraph of Section 17.2.18 needs to be defined.

5. Response to Q 260.12 and FSAR text needs commitment to one year minimum experience in a nuclear QA organization or an acceptable alternative.

6. Response to Q 260.15 which applies Regulatory guides to ASME Code covered items "following receipt at the station" needs to be deleted or justified.

7. The sentence of the response to Q 260.32 which states: "The designation of those activities....." needs to be incorporated into the FSAR text.

8. The response to Q 260.50 needs to be revised to clarify that inspection of operating activities are not performed by personnel within the same group as those performing the activity.

9. The response to Q 260.60 should reference Technical Specification requirements.

10. The response to Q 260.65 should include the commitment incorporated into FSAR Section 17.2.16 that: "For significant conditions adverse to quality not identified by action requests, such as LERs and NRC/INPO/CMAP findings, NQA is involved in the review of such conditions and provides oversight to assure timely follow-up and close out through monitoring, auditing, and commitment verification."

11. The issues below relate to the scope of the operational QA program as described in Tables 3.2-1 and 17.2-1 through 17.2-4.

    a. Information noted for the following items needs to be incorporated into FSAR Table 3.2-1.

        (1) Reactor internal structures, other (I.f) - Show seismic class II/I, note 50, and the last paragraph under item a on page SRAI (1)-12 (Wayne Hodges, X-29410).

        (2) Spent fuel pool liner (XIX.I) - Pertinent provisons of 10 CFR 50 Appendix B should be applied during the operations phase (Kirkwood, X-28436).

(3) Modifications of roof and site drainage systems including such things as drains, parapets, grading, culverts, and channels - Pertinent provisions of 10 CFR 50 Appendix B should be applied during the operations phase (Jachowski, X-28104).

(4) Electro/hydraulic valves of the HPCI and RCIC turbines - Provide a specific reference in Table 3.2-1 to the E/H valves (Thomas, X-29445).

(5) Emergency support facilities display system - This system does not appear to have been added to Table 3.2-1 as indicated in item k on page SRAI (1)-13. Pertinent provisions of 10 CFR 50 Appendix B should be applied during the operations phase (Note 51 acceptable).

b. Table 3.2-1 item XVIII.j, "Shore protection at intake structure" should be clarified to show whether or not both the sheetpile retaining wall and the quarrystone revetments adjacent to the service water intake structure are included (Jachowski, X-28104).

c. Containment isolation valves that are not part of the principal components shown in Table 3.2-1 but that are required per General Design Criteria 54-56 should be subject to the pertinent provisions of 10 CFR 50 Appendix B during the operations phase. Such a commitment in FSAR Table 3.2-1 is required. (Ruth, X-29491).

d. Item on page SRAI (1)-13 states: "The unit vent stacks are Q-listed as shown in revised Table 3.2-1, Item XIX.g." This is acceptable except the table shows "QA Requirements = N" for this item. The stacks should be subject to the pertinent provisions of 10 CFR 50 Appendix B during the operations phase. (Kirkwood, X-28436).

e. Item b on page SRAI (1)-12 addresses the Q-listing of reactor building penetrations. These should be subject to the pertinent provisions of 10 CFR 50 Appendix B during the operations phase (Kirkwood, X-28436).

12. Page 28 of FSAR Table 1.11-1 should be revised to reflect the response to NRC Questions 260.14, 260.29, and 260.55. If this does not result in the deletion of page 28 of the table, the response to the questions listed should be revised such that the page can be deleted.

ATTACHMENT 1: RESPONSE TO NRC OUTSTANDING QA PROGRAM ITEMS (THROUGH AMMENDMENT 5)

Item 1

The commitment to Regulatory Guides in Section 1.8 and on pages 17.2-8 and 17.2-9 does not address the operations phase (see response to Q260.15 and item 8 of the 3/15/84 meeting).

Response

Update of Section 1.8 has been completed and will be transmitted to NRC as part of Amendment 6.

Item 2

The first sentence of the third paragraph of Section 17.2.16 is garbled.

Response

This sentence should read:

For significant conditions adverse to quality not identified by NQA, such as LERs, NRC/INPO/CMAP findings, NQA is involved in the review of such conditions and provides oversight to assure timely follow-up and close out through monitoring, auditing, and commitment verification.

This change will be incorporated in Ammendment 7.

Item 3

The "as applicable" on the third line of page 17.2-29 needs to be deleted or defined.

Response

This change (i.e., to delete "as applicable") will be incorporated in Amendment 7.

Item 4

"Periodically" in the last paragraph of Section 17.2.18 needs to be defined.

Response

The introduction to Section 17.2 (page 17.2-1) states, in part:

To assess the effectiveness of the PSE&G quality assurance program, independent auditors from outside the company periodically audit the program for compliance with 10 CFR 50, Appendix B, and other regulatory commitments. Independent audits shall be conducted at least every two years. Reports of such audits are made directly to upper management.

Section 17.2.18 has also been revised to include this information.

Item 5

Response to Q260.12 does not yet commit to one year minimum experience in a nuclear QA organization.

Response

Section 17.2.1.1.4.1 has been revised to include the information requested.

This change will be incorporated in Amendment 7.

Item 6

Response to Q260.15 which applies Regulatory Guides to ASME Code covered items "following receipt at the station" is unacceptable.

Response

It is felt that the following PSE&G position, as presented to NRC at the 3/15/84 Hope Creek QA Program review meeting, is consistent with good industry practices and hence does not result in a compromise of controls affecting safety.

PSE&G will comply with applicable supplemental ANSI standards during procurement of Section III components to the extent that such ANSI standards are contained in the applicable ASME edition which the supplier has in effect at the time of procurement.

It should be noted that NCA-4000, 1983 edition, endorses the 18 criteria of NQA-1, 1979 for N, NV, NPT and NA certificate holders for class 1, 2, 3, MC, CS, CB and CC construction. Therefore, elements of the 10 CFR 50, Appendix B program, not currently included in the ASME QA program for suppliers of the above code items, will be addressed as each supplier updates the latest edition of the code.

Item 7

The sentence of the response to Q260.32 which states "the designation of those activities..." should be incorporated into the FSAR text.

Response

The requested information will be incorporated in Section 17.2.5 and included in Amendment 7.

Item 8

The response to Q260.50 needs to be revised per page 24 of the 3/15/84 meeting summary.

Response

The information requested was incorporated in Section 17.2.10 (page 17.2-27) of Amendment 5. In addition, response to Question 260.50 will be ~~revised to reflect this update.~~ corrected to reference 17.2.10 The latter revision will be included in Amendment 7.

Item 9

The response to Q260.60 should reference Technical Specification requirements per page 26 of the 3/15/84 meeting summary.

Response

Response to Question 260.60 has been revised to include the requested information. This revision will be included in Amendment 7.

Add amended response (i.e., "In addition, administrative procedures ... are provided") to FSAR text.

Item 10

The response to Q260.65 should include commitment per page 27 of the 3/15/84 meeting summary.

Response

As stated in response to Question 260.65, the information has been included in Section 17.2.16 (page 17.2-32) as part of Amendment 5. In addition, minor revisions to this information, as previously described in response to Item 2, will be included in Amendment 7.

Item 11

The issues below relate to the scope of the operational QA program as described in Tables 3.2-1 and 17.2-1 through 17.2-4 and as responded to per pages 33 through 36 (item 29) of the 3/15/84 meeting summary:

(a) The staff position is that the following items should have the pertinent QA requirements of 10 CFR 50, Appendix B applied during the operational phase.

(1) feedwater spargers and other reactor internal structures

(2) spent fuel pool liner

(3) modifications of roof and site drainage systems including such things as drains, parapets, grading culverts and channels

(4) electro/hydraulic valves of the HPCI and RCIC turbines

(5) emergency support facilities display system

(b) Clarify that Table 3.2-1 item XVIII.j, "Shore protection at intake structure" includes both the sheetpile retaining wall and the quarrystone revetments adjacent to the service water intake structure.

(c) Provide a commitment in Table 3.2-1 that containment isolation valves that are not part of the principal components shown in Table 3.2-1, but that are required per GDC 54-56 will be subject to the pertinent provisions of 10 CFR 50 Appendix B.

Response

TAB

FSAR CHANGES

The PSE&G policies and organization structure assure that the manager - quality quality assurance nuclear operations has sufficient organizational freedom and independence to carry out his responsibilities.

17.2.1.1.4.1 Nuclear Operations Quality Assurance Personnel Qualifications

and must be obtained within the quality assurance organization

The manager - NQA and engineers reporting directly to him must each have a combination of 6 years of experience in the field of quality assurance and operations. At least 1 of these 6 years of experience must be in the overall implementation of a nuclear power plant quality assurance program. A minimum of 1 year and a maximum of 4 of the 6 years of experience may be fulfilled by related technical or academic training. Personnel performing inspections, examinations, and test activities are certified as Level I, Level II, Level III as appropriate to their responsibilities, also in accordance with Regulatory Guide 1.58, as noted.

The manager - nuclear operations quality assurance fulfills the above qualifications with the addition of the following:

a. Knowledge and experience in quality assurance,

b. High level of leadership with the ability to command the respect and cooperation of company personnel, vendors, and construction forces

c. Initiative and judgment to establish related policies to attain high achievements and economy of operations.

17.2.1.1.5 Independent Review Groups

Three advisory groups are responsible for reviewing and evaluating items related to nuclear safety. The overall responsibilities of these groups are included in the following sections. More detailed descriptions are contained in Section 13.4.

The SORC is an in station advisory group. Composed of key station personnel, its responsibilities include review of plant and Maintenance Department personnel who perform visual inspection as part of the Inservice Inspection program.

Personnel requiring certification are evaluated to establish their qualifications for their respective level and discipline. Recertification is based upon demonstrated continued proficiency or requalification, if necessary. Personnel requiring certification in accordance with Regulatory Guide 1.58 are limited to NQA personnel who perform inspection and test activities, ~~and~~ members of the Operational Test Group (OTG) who perform post-design modification testing. ~~NQA and OTG and calibration~~ These personnel receive a periodic training needs assessment to identify additional supportive training needs as well as to evaluate individual post-training performance. The assessment period is three years or less. Inspection and test activities not requiring personnel certification per Regulatory Guide 1.58 include Technical Specification surveillances and periodic inspection and test of fire protection equipment. These personnel are qualified and retrained in accordance with applicable requirements of Regulatory Guide 1.8.

Training programs of supporting organizations are described in their manuals, which are required to comply with the quality assurance program.

The Nuclear Training Center is responsible for the licensed operator training and retraining, in addition to other technical and supervisory training programs, including General Employee Indoctrination, which is required for all personnel having access to the station.

17.2.3 DESIGN CONTROL

The design control program includes activities such as field design engineering, associated computer programs, compatibility of materials, and accessibility for inservice inspection, maintenance, and repair.

During the operations phase, issuance of new drawings and revisions to existing drawings require the implementation of a design change.

The nuclear support division procedures, approved by the manager - nuclear operations QA, provide implementation guidance for the intent of Regulatory Guide 1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants." Within that division, the nuclear engineering section has the following responsibilities:

The scope of the design control programs includes design activities associated with the preparation and review of design documents, including the correct translation of applicable regulatory requirements into design modification, procurement and procedural documents.

The designation of those activities requiring detailed procedures is made by cognizant department heads and as a minimum, complies with applicable requirements of Regulatory Guide 1.33.

c. Provide right of access for source surveillance and audit by NQA or its agents

d. Provide for required supplier documentation to be submitted to PSE&G or maintained by the supplier, as appropriate

e. Provide for PSE&G review and approval of critical procedures prior to fabrication, as appropriate.

Procurement documents require suppliers and contractors of other than commercial grade items to provide services or components in accordance with a quality assurance program that complies with applicable parts of 10 CFR 50, Appendix B. The requirement for notifying PSE&G of procurement requirements that have not been met is conveyed to the supplier through the standard warranty provision contained in each Purchase Order. In addition, where 10 CFR 21 is imposed, suppliers are required to comply with applicable reporting requirements.

17.2.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Organizations engaged in Q- and F-designated activities are required to perform these activities in accordance with written and approved procedures, instructions, or drawings, as appropriate.

Simple routine activities that can be performed by qualified personnel with normal skills do not require a detailed written procedure. Complex activities require detailed instructions. ~~An activity is defined as complex upon the designation of the responsible department manager~~.

Procedures include, as appropriate, scope, statement of applicability, references, prerequisites, precautions, limitations, and checkoff lists of inspection requirements, in addition to the detailed steps required to accomplish the activity. Instructions, procedures, and drawings also contain acceptance criteria where appropriate.

The general manager - Hope Creek operations is responsible for assuring that station procedures are prepared, approved, and implemented in compliance with the station administrative procedures. Documents affecting nuclear safety are reviewed by the station operations review committee (SORC) for technical content, by NQA for quality assurance requirements, and are approved by the responsible station department manager or his designee.

The general manager - nuclear support is responsible for issuing specifications, drawings, blueprints, and instruction and technical manuals associated with Q- and F-designated structures, systems, and components. Approved and implemented modifications and design changes are incorporated to these reference documents for the life of the station. Master lists of current editions or revisions of these documents are periodically issued by the general manager - nuclear support to the general manager - Hope Creek operations to periodically assure that only current and approved referenced documents are used at the station.

NQA reviews and approves station inspection plans and procedures that implement the quality assurance program, including testing, calibration, maintenance, modification, rework and repair. Changes to these documents are also reviewed and approved. In addition, NQA is responsible for review and approval of PSE&G specifications, test procedures, and results of testing.

17.2.6 DOCUMENT CONTROL

Instructions, procedures, drawings, and changes thereto are reviewed for inclusion of appropriate quality assurance requirements and are approved by apppropriate levels of management of the PSE&G organizations producing such documents, and distributed on a timely basis to using locations. Measures are provided for the timely removal of obsoleted or superseded documents from the using location. Supplier documents are controlled according to contractual agreements with suppliers.

The following is a generic minimum listing of key documents for the operational phase, showing organization responsibility for review and (or) approval, including changes thereto:

a. Design specification - nuclear support, ~~department,~~ NQA

b. Design modification ~~manufacturing, construction, and installation drawings~~ - nuclear ~~department~~ support, nuclear services Hope Creek operations, NQA

c. Procurement documents - initiating nuclear department organization, purchasing department, nuclear services, NQA

d. Nuclear department manual - ~~Quality assurance manual - NQA~~ nuclear department organizations responsible for implementation, NQA

e. Station administrative procedures - ~~general manager~~ Hope Creek operations, NQA

f. Maintenance, modification, and calibration procedures for Q- and F-designated station work activities - ~~general manager~~ Hope Creek operations, NQA

g. Operating procedures - ~~general manager~~ Hope Creek operations, SORC

h. FSAR - nuclear ~~department~~ services, ~~NQA~~ and other nuclear department organizations responsible for implementing applicable sections.

i. Maintenance, inspection, and testing instruction - nuclear department implementing organizations, NQA

j. ~~Q listed test procedures - nuclear department, NQA~~ Post modification test procedures - nuclear services, NQA

~~k. Design change requests - nuclear department, NQA~~

In addition, NQA involvement in the work activity includes a review of nonsafety-related work orders for proper classification prior to conducting the activity and a review of completed safety-related work orders.

The establishment and maintenance of a document control system for all instructions, procedures, specifications, and drawings received from the nuclear department, or prepared at the station for use in operating, maintaining, refueling, or modifying items and services covered by the quality assurance program, is the responsibility of the general manager - Hope Creek operations. The administrative procedures manual describes the control of specific documents. Control of station practices is included in the administrative procedures and in department directives authorized by the responsible station department managers. Measures are established to assure that the administrative procedures and department directives are up-to-date, are properly

e. Critical test sequence

f. Acceptance criteria.

Test results, including verification of above items, ~~as applicable~~ are documented and reviewed for acceptability by the qualified department representative. System tests performed following modifications to Q- and F-designated systems require review of test procedures and test results by the SORC.

NQA ~~maintains~~ monitors ~~ing over~~ the conduct of ~~the design change~~ post modification ~~acceptance~~ tests to assure compliance with the test procedure. Test results are reviewed for the following:

a. Presentation of proper documentation

b. Assurance that tests meet objectives

c. Identification and reporting of unacceptable results and initiation of corrective measures.

17.2.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Test equipment, instrumentation, and controls used to monitor and measure activities affecting quality and personnel safety are identified, controlled, and calibrated at specific intervals by cognizant nuclear department personnel. Written procedures for meeting these requirements include provisions for:

a. Specifying calibration frequency

b. Recording and maintaining calibration records

c. Controlling and calibrating primary and secondary standards

d. Determining methods of calibration

e. Tracing use on safety-related items.

repair or "use-as-is" are required to be approved by the responsible engineering representative. Rework or repair of nonconforming material, parts, or components is inspected and/or retested in accordance with specified test and inspection requirements established by the cognizant engineer, based on applicable ~~code~~ requirements.

NQA and the nuclear department review nonconformance reports for quality problems, including adverse quality trends, and initiate reports to higher management, identifying significant quality problems with recommendations for appropriate action.

17.2.16 CORRECTIVE ACTION

Organizations involved in activities covered by the quality assurance program are required to maintain corrective action programs commensurate with their scope of activity. Noncompliances with the quality assurance program identified by NQA are documented and controlled by issuing an action request. NQA reviews responses to action requests for adequacy and monitors these action requests through periodic summary and status reports to management.

Responses to action requests are based on the four elements of corrective action, which are:

a. Identification of cause of deficiency

b. Action to correct deficiency and results achieved to date

c. Action taken or to be taken to prevent recurrence

d. Date when full compliance was or will be achieved.

For significant conditions adverse to quality not identified by NQA, such as LERs, NRC/INPO/CMAP findings, NQA is involved in the review of such conditions and provides oversight to assure timely follow-up and close out through monitoring, auditing, and commitment verification.

Items 3 and 4 are optional for noncompliances that do not have a significant effect on the quality assurance program.

d. Indoctrination and training

e. Implementation of operating and test procedures

f. Calibration of measuring and test equipment

g. Fire protection

h. Other applicable activities delineated in Table 17.2-2.

The audit data is analyzed and a written report of the results of each audit is distributed to appropriate management representatives of the organization(s) audited, as well as to other affected management personnel. Included in the report is a statement of QA program effectiveness. ~~Periodically.~~ NQA is audited by independent auditors at least every two years to verify implementation of the corporate quality assurance program. Reports of these audits are directed to appropriate PSE&G management personnel.

QUESTION 260.50 (SECTION 17.2)

Describe the provisions which assure that when inspections associated with normal operations of the plant (such as routine maintenance, surveillance, and tests) are performed by individuals other than those who performed or directly supervised the work, but are within the same group, the following controls are met: (SRP Section 17.2.10, item 2)

a. The quality of the work can be demonstrated through a functional test when the activity involves breaching a pressure retaining item.

b. The qualification criteria for inspection personnel are reviewed and found acceptable by QANO prior to initiating the inspection.

RESPONSE

See response to Questions 260.7 and 260.19.

Section 17.2.10 ~~(Page 17.2-25)~~ has been revised to provide additional information requested.

QUESTION 260.60 (SECTION 17.2)

Describe those provisions which assure that procedures are established to control altering the sequence of required tests, inspections, and other safety-related operations. Such actions should be subject to the same controls as the original review and approval. (14.3)

RESPONSE

Section 17.2.11 states in part:

Test procedures prescribe, as applicable:

(d) Critical test sequence

.....Test results are documented and reviewed for acceptability by the qualified department representative.

In addition, station administrative procedures provide for the use of temporary changes. Detail instructions for implementation of temporary changes are provided.

which are controlled in accordance with Technical Specifications

TAB

SECTION 1.8, AMENDMENT 6

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

1.8.1 NON-NSSS ASSESSMENT OF CONFORMANCE

The extent of non-nuclear steam supply system (NSSS) compliance with the NRC Regulatory Guides is indicated here and, where applicable, reference is made to the final safety analysis report (FSAR) section(s) that describe the appropriate design feature.

Determination of conformance is based on a comparison of the Hope Creek Generating Station (HCGS) non-NSSS design and construction to the latest version of the Regulatory Guides. Variances are discussed and justified in this section where the design deviates from regulatory guidelines, or where compliance has been qualified by an interpretation of the Regulatory Guide. Positions stated with respect to Regulatory Guide compliance will apply during the operations phase unless otherwise stated. In general, the statement, "although Regulatory Guide 1.XXX does not apply to HCGS, per its implementation section..." applies only during construction and startup phase; i.e., the Regulatory Guide is applicable during the operations phase. CN30'

1.8.1.1 Conformance to Regulatory Guide 1.1 (Safety Guide 1) Revision 0, November 2, 1970: Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps

HCGS complies with Regulatory Guide 1.1, as described below.

The suction piping for all pumps required for safe shutdown of the reactor, during both normal and accident conditions, including the cooling of both the core and the containment, is designed and located to ensure adequate net positive suction head (NPSH). The available NPSH for the residual heat removal (RHR) and core spray pumps is based on a torus water temperature of 212°F, with the pool surface at 14.7 psia. The calculated available NPSH for the high pressure coolant injection (HPCI) pump is based on a water temperature of 170°F, with the pool surface at 14.7 psia.

For further discussion, see Sections 5.4.7, 6.2.2, and 6.3.2.

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.2 Conformance to Regulatory Guide 1.2, (Safety Guide 2) Revision 0, November 2, 1970: Thermal Shock to Reactor Pressure Vessels

HCGS complies with Regulatory Guide 1.2, as described below:

An investigation of the structural integrity of boiling water reactor (BWR) pressure vessels during a design basis accident (DBA) determined that, based on the methods of fracture mechanics, failure of the vessel by brittle fracture does not occur as a result of a DBA.

See Section 5.3 for further discussion of fracture toughness of the reactor pressure vessel (RPV) and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.3 Conformance to Regulatory Guide 1.3, Revision 2, June 1974: Assumptions Used For Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident For Boiling Water Reactors

HCGS complies with Regulatory Guide 1.3. See Chapter 15 for discussion of accident analyses.

1.8.1.4 Conformance to Regulatory Guide 1.4, Revision 2, June 1974: Assumptions Used For Evaluating The Potential Radiological Consequences of a Loss of Coolant Accident For Pressurized Water Reactors

Regulatory Guide 1.4 deals with pressurized water reactors (PWRs) only and is, therefore, not applicable to HCGS.

1.8.1.5 Conformance to Regulatory Guide 1.5 (Safety Guide 5), Revision 0, February 1, 1971: Assumptions Used for Evaluating The Potential Radiological Consequences of a Steam Line Break Accident For Boiling Water Reactors

HCGS complies with Regulatory Guide 1.5. See Chapter 15 for discussion of accident analyses.

due to the excitation current inrush while the transformers are energized and lasts for approximately six cycles. The first motor load applied in the RHR motor, after closure of the SDG circuit breaker. The RHR motor circuit breaker has a closing permissive from the bus undervoltage relays. With the current setting of these relays (set to dropout at 70 percent and to pickup at 78 percent) the RHR motor circuit breaker will close when permitted. It takes 4.5 cycles for this circuit breaker to close. During this interval the generator has recovered its voltage in excess of 90 percent. This will be verified during the preoperational tests described in Section 14.2.12.1.30.

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No change from Amend.

Compliance with Position C.6 of Regulatory Guide 1.9 is discussed in Section 1.8.1.108.

For further discussion of onsite power systems, see Section 8.3.

1.8.1.10 Conformance to Regulatory Guide 1.10, Revision 1, January 2, 1973: Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

Although Regulatory Guide 1.10 was withdrawn by the NRC on July 21, 1981, HCGS complies with it.

The original Cadweld testing program in the preliminary safety analysis report (PSAR) was based on using only sister splices. The program was later revised before the start of construction to conform with the Regulatory Guide using a combination of production and sister splices. When newer technical criteria for Cadwelding developed, the architect-engineer revised the program to delete the tensile test frequency requirements for each splicing crew. The new criteria conformed to the requirements of ANSI N45.2.5, 1978, as endorsed by Regulatory Guide 1.94. However, the letter dated August 5, 1981, NRC to PSE&G, from R. L. Tedesco to R. L. Mittl, requested that the sample frequency requirements of this guide be implemented. Since November 30, 1981, HCGS has been in complete compliance with this Regulatory Guide.

For further discussion, see Section 3.8.6.

1.8.1.11 Conformance to Regulatory Guide 1.11 (Safety Guide 11), Revision 0, February 1, 1971: Instrument Lines Penetrating Primary Reactor Containment

HCGS complies with Regulatory Guide 1.11, except as noted below.

Containment pressure sensing lines are not provided with an automatic or remotely operated isolation valve as specified in Position C.1.c of Regulatory Guide 1.11. Sensing lines are not isolated automatically upon a containment isolation signal because the pressure sensors provide a reactor protection system (RPS) signal. The capability for remote operation is not useful to the operator because remote indication of failure of a specific line is not available. However, these lines are provided with manual isolation valves for local operation and are checked for leakage during normal instrumentation calibrations.

For further discussion of containment isolation provisions, see Section 6.2.4.

1.8.1.12 Conformance to Regulatory Guide 1.12, Revision 1, April 1974: Instrumentation for Earthquakes

HCGS complies with ANSI N18.5-1974, as endorsed and modified by Regulatory Guide 1.12, subject to the clarification that the response-spectrum recorders required by Paragraph C.1.c are not supplied as discrete instruments. Instead, triaxial time-history accelerographs are provided, at the required locations, with a multichannel magnetic tape recorder and a response-spectrum analyzer. This system provides more complete information than that presented by response-spectrum recorders.

For further discussion, see Section 3.7.4.

1.8.1.13 Conformance to Regulatory Guide 1.13, Revision 1, December 1975: Spent Fuel Storage Facility Design Basis

HCGS complies with Regulatory Guide 1.13 with the following exception:

1.8.1.23 Conformance to Regulatory Guide 1.23 (Safety Guide 8), Revision 0, February 17, 1972: Onsite Meteorological Programs

HCGS complies with Regulatory Guide 1.23.

1.8.1.24 Conformance to Regulatory Guide 1.24 (Safety Guide 24), Revision 0, March 23, 1972: Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Regulatory Guide 1.24 is not applicable to HCGS.

1.8.1.25 Conformance to Regulatory Guide 1.25 (Safety Guide 25), Revision 0, March 23, 1972: 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

HCGS complies with Regulatory Guide 1.25.

1.8.1.26 Conformance to Regulatory Guide 1.26, Revision 3, February 1976: Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

HCGS complies with Regulatory Guide 1.26, with the clarifications outlined below. CN305

PSE&G's position is that equipment that is important to safety is safety-related and therefore does not distinguish between these terms. PSE&G does recognize the need for the assurance of the specified operation of certain non-safety-related structures, systems and components, such as fire protection systems, radioactive waste treatment, handling and storage systems, and Seismic Category II/I items. Such assurance is documented through the specification of limited quality assurance programs (described in Table 3.2-1, footnotes (22), (50) and (52). In addition, items designated "D+" in Table 3.2-1 will be included in the QA program during operations.

The exception to Position C.2.b is that since the reactor recirculation pumps do not perform any safety function and since failure of the reactor coolant pumps due to seal or cooling water failure does not have serious safety implications, the control rod drive (CRD) seal purge supply and reactor auxiliaries cooling system (RACS) cooling water to the seal coolers are quality group D.

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Additionally, Position C.2.b of Regulatory Guide 1.26 requires that cooling water systems important to the safety function of the standby diesel generators be Quality Group C. HCGS's diesel generator cooling water systems are classified as Quality Group C except for the engine mounted piping systems (such as the lube oil headers, water headers, cylinder heads, etc). The engine mounted piping systems are part of the diesel engine and its auxiliary support sytems which, as stated in Section B of the Regulatory Guide, are not covered by this guide. These systems are manufactured to the manufacturer's proprietary design requirements which do not necessarily meet the requirements of ASME Section III or ANSI B.31. However, the components used are pressure tested and the manufacturing processes are monitored as a part of the suppliers approved QA program, which addresses the 18 criteria contained within 10 CFR 50, Appendix B.

Additional quality assurance requirements invoked by the applicant include:

a. periodic documented subsupplier audits (including plant visits),

b. review and approval of subsupplier QA programs and manuals,

c. test and inspection audits,

d. calibration of test gauges before and after use, and

e. control of calibration records and acceptance devices.

With the imposition of the above design, manufacturing, and testing controls, the on-skid and off-skid piping and components have been made to be equivalent to Quality Group C. This meets the requirements in Section B of the guide to design, fabricate, erect and test the diesel engine and its auxiliary support systems to quality standards commensurate with the safety function to be performed.

NUREG-0737, Item II.k.3.25 extends the requirements of Position C.2.b by requiring demonstration that the consequences stemming from a loss of cooling water to the reactor recirculation pump seal coolers is acceptable following a loss of power for at least 2 hours. NEDO-24951 (Reference 5.4-4) confirms that the HCGS design meets the requirements of NUREG-0737, Item II.k.3.25.

See Section 3.2.2 for further discussion and Section 1.8.2 for NSSS assessment of this Regulatory Guide.

1.8.1.27 Conformance to Regulatory Guide 1.27, Revision 2, January 1976: Ultimate Heat Sink For Nuclear Power Plants

HCGS complies with Regulatory Guide 1.27. The ultimate heat sink (UHS) is the Delaware River, which is a large, single water source as defined by the Regulatory Guide. The service water equipment required for the dissipation of residual heat is all safety-related and redundant, with the exception of the service water discharge piping outside of the reactor building. This piping normally discharges into the circulation water system (CWS). However, if some natural or site-related event occurs and blocks the flow, there are rupture discs in the safety-related portion of the service water discharge piping that allow the water to be safely diverted onto and across the lower yard surface area, thus completing the cooling loop between the UHS and the plant.

For further discussion of the station service water system (SSWS) and the UHS, see Sections 9.2.1 and 9.2.5.

1.8.1.28 Conformance to Regulatory Guide 1.28, Revision 2, February 1979: Quality Assurance Program Requirements (Design and Construction)

Although Regulatory Guide 1.28, Revision 2, is not applicable to HCGS, per its implementation section, HCGS complies with ANSI N45.2-1977 as modified and interpreted by Regulatory Guide 1.28. The architect-engineer quality program for the design and construction of safety-related items meets the requirements of

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this standard, subject to clarifications and interpretations stated below.

The introduction of Position C of Regulatory Guide 1.28 states that ANSI N45.2-1977 provides an adequate basis for complying with the quality assurance program requirements of Appendix B of 10 CFR 50, subject to the listed provisions of the Regulatory Guide. The architect-engineer interprets the scope of the endorsement of N45.2-1977 by Regulatory Guide 1.28, to be limited to the safety-related structures, systems, and components covered by Appendix B to 10 CFR 50.

Nothing in Position C.1 of Regulatory Guide 1.28 requires the PSAR or FSAR to be a design document, and it should not be construed to be a design document. It is a description document, and its preparation is covered by parts of 10 CFR 50 other than Appendix B and Regulatory Guides other than Regulatory Guide 1.28.

Position C.2.a is not followed because provisions indicated by the verb "should", of ANSI N45.2-1977 (on procedures for critical, sensitive, or perishable articles and on use and control of special handling tools) are treated by the architect-engineer as guidelines.

Nothing in Position C.4 is construed to preclude the extension of an ASME B&PV Code quality assurance program to non-Code items where the Code quality assurance program element adequately covers the programmatic activity performed. A Code quality assurance program could be extended to cover functional operability without invoking ANSI N45.2. The criteria regarding functional operability are technical in nature and, as such, are controllable under either an ASME B&PV Code or ANSI N45.2 quality assurance program.

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

This Regulatory Guide is not applicable during the operations phase. CN305

1.8.1.29 Conformance to Regulatory Guide 1.29, Revision 3, September 1978: Seismic Design Classification

HCGS complies with Regulatory Guide 1.29, except as noted below.

Position C.1.e of Regulatory Guide 1.29 requires that those portions of the steam systems of boiling water reactors extending from the outermost primary containment isolation valve up to but not including the turbine stop valve and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation be designated as Seismic Category I and be designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional. This position also requires that the pertinent quality assurance requirements of Appendix B to 10 CFR 50 be applied to all activities affecting the safety-related functions of these systems and components. Additionally, the turbine stop valve should be designed to withstand the SSE and maintain its integrity.

The main steam line classification and design is based on the approach discussed in Standard Review Plan 3.2.2, Revision 1, July 1981, Appendix B. The main steam lines (MSL) from the second isolation valve up to and including MS stop valve and all the branch lines 2 1/2-inches in diameter and larger between these two valves up to and including the first valve in the branch line is classified under quality groug B (ASME Section III, Class 2). The main steam line piping between MS stop and the turbine main stop valve is ASME Section III, Class 3 instead of D classification as required in Appendix B. This portion of MSL is not classified as safety-related, is not specifically designed to Seismic Category I standards, and is not housed in Seismic Category I structures, as discussed in Appendix B. This different approach satisfies SRP 3.2.2 acceptance criteria requirements and results in an acceptable level of safety.

Editorial

Position C.1.h requires that cooling water and seal water systems or portions of these systems that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps, be designated and designed as Seismic Category I systems and components.

The CRD seal purge and seal cooling from the RACS for the reactor recirculation pumps are not designed to withstand an SSE, as the reactor recirculation pumps do not perform any safety function, and failure does not have serious safety implications. NUREG-0737, Item II.k.3.25 extends the requirements of Position C.1.h by requiring demonstration that the consequences stemming from a loss of cooling water to the reactor recirculation pump seal coolers is acceptable following a loss of ac power for at least 2 hours. NEDO-24951 (Reference 5.5-4) confirms that the HCGS design meets the requirements of NUREG-0737, Item II.k.3.25. NEDO-24083 (Refence 1.8-2) shows that if the seal and cooling systems to the reactor recirculation pump fail to operate, the leakage past the recirculation pump seal is sufficiently small so that no safety concerns exist.

Position C.2 of Regulatory Guide 1.29 requires that items that would otherwise be classified non-Seismic Category I, "but whose failure could reduce the functioning" of the items important to safety "to an unacceptable safety level," are to be "designed and constructed so that the SSE would not cause such failure." In addition, Position C.4 of Regulatory Guide 1.29 requires that the pertinent quality assurance requirement of Appendix B to 10 CFR 50 be applied to the safety requirements of such items. Both these requirements are considered to be adequately met by establishing the following practices to such items: CN30

a. During the construction and operations phase, design and design control for features of such items that should not fail are carried out in the same manner as for items directly important to safety. This includes the performance of appropriate design reviews.

b. During the construction phase, field work is performed under the direction of experienced field construction superintendents and is inspected by quality control engineers stationed at the site. The quality control engineers are responsible for verifying that construction is performed in accordance with the design drawings and specifications and with applicable standard codes and specifications.

c. During the construction phase, such items are neither purchased to a code higher than normal system design dictates, nor is the quality assurance program of 10 CFR 50, Appendix B, applied to their procurement. However, these items are identified in the applicable documents. During the operations phase applicable procurement documents, design modifications documents, and station work orders will be reviewed by nuclear operations quality assurance (NQA) for designation of appropriate quality assurance controls.

Position C.3 of Regulatory Guide 1.29 requires that Seismic Category I design requirements be extended "to the first seismic restraint beyond the defined boundaries." Since seismic analysis of a piping system necessitates division of the systems into discrete segments terminated by fixed points, the seismic design cannot be terminated at a seismic restraint. However, it is extended to the first point in the system that can be treated as an anchor to the plant structure. In addition, Position C.4 of Regulatory Guide 1.29 requires that the pertinent quality assurance requirement of Appendix B to 10 CFR 50 be applied to the safety requirements of such items. Both these requirements are considered to be met adequately by establishing the following practices:

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a. During the construction and operations phase, design and design control for such items are carried out in the same manner as for items directly important to safety. This includes the performance of appropriate design reviews.

b. During the construction and operations phase, walk-through inspections are performed by representatives of the originating design group (nuclear engineering department during the operations phase) to ensure that the final installation of such items is in accordance with documents that formed the basis for the seismic analysis of the items.

c. During the construction phase, such items are neither identified as requiring the quality assurance requirements of 10 CFR 50, Appendix B, nor purchased to a code higher than normal system design dictates. During the operations phase, applicable procurement documents, design modification documents and station work orders will be reviewed by NQA for designation of appropriate quality assurance controls.

See Section 3.2.1 for further discussion of seismic design classification and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.30 Conformance to Regulatory Guide 1.30 (Safety Guide 30), Revision 0, August 11, 1972: Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

HCGS complies with Regulatory Guide 1.30.

See Section 17.2 for further discussion of quality assurance and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

See Section 6.1.2 for further discussion of ESF materials and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.55 Conformance to Regulatory Guide 1.55, Revision 0, June 1973: Concrete Placement in Category I Structures

Regulatory Guide 1.55 was withdrawn by the NRC on July 21, 1981. However, the placement of concrete in Seismic Category I structures is in accordance with this Regulatory Guide, with the exceptions discussed below.

Positions C.2 and C.3 state the presumed functional responsibilities of the "designer" and the "constructor." The designer's role includes the responsibilities of checking shop drawings and locations of construction joints. For HCGS, the former is fully delegated to the qualified architect-engineer-constructor although the architect-engineer design engineering office may check significant portions and may advise construction accordingly. The responsibility for construction joint location is partially delegated to the field in the sense that the field must follow the guidelines set out in the design drawings and specifications prepared by engineering.

For further discussion of the design of Seismic Category I structures, see Section 3.8.

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This Regulatory Guide is not applicable during the operations phase.

1.8.1.56 Conformance to Regulatory Guide 1.56, Revision 1, July 1978: Maintenance of Water Purity in Boiling Water Reactors

HCGS complies with Regulatory Guide 1.56, with exception to Position C.4.c. This exception is discussed in Section 10.4.6.2.1.

For further discussion, see Sections 5.2.3, 5.4.8, and 10.4.6.

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.57 Conformance to Regulatory Guide 1.57, Revision 0, June 1973: Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

HCGS complies with Regulatory Guide 1.57, except that the loading combinations and stress limits of Position C.1.b(2) are not used. The loading combinations and stress limits that are used by HCGS in the analysis of the primary containment during a postulated post-LOCA flooded condition are recognized by Standard Review Plan Section 3.8.2, Paragraph II.3.b.iii.e.

1.8.1.58 Conformance To Regulatory Guide 1.58, Revision 1, September 1980: Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

HCGS complies with Regulatory Guide 1.58.

The architect-engineer indicates that the requirements of ANSI N45.2.6-1978, as modified and interpreted by Regulatory Guide 1.58, are met by its quality program for safety-related items.

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During the operations phase, PSE&G complies with Regulatory Guide 1.58, Revision 1 with an exception to Position C.6, which requires a high school diploma or an earned General Education Development equivalent. ANSI N45.2.6-1978, Paragraph 3.5, recognizes that other factors may provide reasonable assurance, i.e., previous performance, satisfactory completion of testing, that a person can competently perform a particular task. These factors will be considered when evaluating education and experience requirements for certification.

See Chapter 17 for further discussion of quality assurance and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.59 Conformance to Regulatory Guide 1.59, Revision 2, August 1977 with Errata Sheet July 30, 1980: Design Basis Floods for Nuclear Power Plants

Although Regulatory Guide 1.59 does not apply to HCGS, per its implementation section, HCGS complies with it.

1.8.1.66 Conformance to Regulatory Guide 1.66, Revision 0, October 1973: Nondestructive Examination of Tubular Products

Regulatory Guide 1.66 was withdrawn by the NRC on September 28, 1977.

See Section 5.2.3 for further discussion of testing on mechanical components and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.67 Conformance to Regulatory Guide 1.67, Revision 0, October 1973: Installation of Overpressure Protection Devices

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Regulatory Guide 1.67 is not applicable to HCGS because there are no open discharge lines where reaction forces are considered to be significant.

1.8.1.68 Conformance to Regulatory Guide 1.68, Revision 2, August 1978: Initial Test Programs for Water-Cooled Nuclear Power Plants

HCGS complies with Regulatory Guide 1.68, with the exceptions and clarifications discussed below.

Position C.1 provides the criteria for selection plant features that are tested during the initial test program. At HCGS, testing is conducted on structures, systems, components, and design features as described in Section 14.2, based on their safety-related functions.

See Section 3.9.2 for further discussion of dynamic testing and analysis.

The objective of Regulatory Guide 1.68 is to describe the scope and depth of a test program, as required, to ensure that plant structures, systems, and components perform satisfactorily in service. The basis for this Regulatory Guide is Appendix B to 10 CFR 50, which specifically applies only to testing the performance of safety-related functions. Therefore, this Regulatory Guide is applied only to plant structures, systems, and components that have safety-related functions, defined as those plant features necessary to ensure the integrity of the RCPB, the capability to shut down the reactor and maintain it in a safely shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in offsite exposures comparable to the guideline exposure of 10 CFR 100.

Safety-related structures, systems, and components are identified as such in Chapter 14 and are tested to meet the requirements of Regulatory Guide 1.68. Other systems and components within the plant that are not safety-related may or may not be tested in accordance with the Regulatory Guide. Since the plant units that are not safety-related by definition do not compromise the safety-related aspects of the plant, it is not planned to test them to the Regulatory Guide.

Regulatory Position C.7 and Section 1.h of Appendix C state that one of the objectives of the initial test program is to verify by trial use that the facility operating and emergency procedures are adequate. Because preoperational test procedures are intended to demonstrate system design criteria, they are conducted under system configurations and conditions different than those required by facility operating and emergency procedures. Therefore, operating and emergency procedures are proven independent of the preoperational test procedures.

Section 1 of Appendix A states that system vibration, expansion, and restraints may be verified by observation as allowed during power-ascension testing by Section 5.0.0 of Appendix A. This

1.8.1.87 Conformance to Regulatory Guide 1.87, Revision 1, June 1975: Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)

Regulatory Guide 1.87 is not applicable to HCGS.

1.8.1.88 Conformance to Regulatory Guide 1.88, Revision 2, October 1976: Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records

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During the operations phase, HCGS complies with ANSI N45.2.9-1974, as modified and interpreted by Regulatory Guide 1.88. During the construction and startup phase compliance is subject to the following specific changes.

The architect-engineer indicates that the original HCGS project commitment, via the Bechtel nuclear quality assurance manual (NQAM), was to ANSI N45.2.9 (Draft 11, Revision 0, January 17, 1973) rather than to ANSI N45.2.9-1974. The NQAM was revised to reference the 1974 document, as modified and interpreted by the guide, subject to the following specific changes:

a. ANSI Section 2.1, Quality Assurance Record System - Add the following sentence at the end of this section: "The procedures shall include control of records required during completion of the work activity."

b. ANSI Section 2.2.2, Nonpermanent Quality Assurance Records - Revise this section to read: "Nonpermanent records are those required to show evidence that an activity was performed in accordance with the applicable requirement but need not be retained for the life of the item and do not meet the criteria listed in Section 2.2.1."

c. ANSI Section 3.2.2, Index - Revise this section to read: "The quality assurance records shall be listed in an index. The index shall include, as a minimum, record retention times and the location of the records within the record system. The index system used by organizations for the retention of quality assurance records shall provide information which can be used to identify the equipment or material."

d. ANSI Section 3.2.7, Retention of Records - Appendix A, referenced in this section, is revised to list only lifetime quality assurance records. This section is revised to read: "Lifetime quality assurance records are listed in Appendix A of this standard. It is recognized that the nomenclature of these records may vary. The organization responsible shall establish in writing the retention times of records not listed in Appendix A."

e. ANSI Section 5.6, Facility - Revise this section to read: "Quality assurance records shall be stored in facilities constructed and maintained in a manner which minimizes the risk of damage or destruction from fire, flooding, or other natural disasters; from the dangers of extreme temperature and humidity; and from infestation by insects and rodents."

There are three satisfactory alternatives for pro iding record storage integrity:

1. Provide dual storage facilities at separate remote locations that are not exposed to a simultaneous hazard

2. Provide a single record storage facility. Where a single record storage facility is maintained, the following features are to be considered in its construction:

   (a) Reinforced concrete, concrete block, masonry, or equal construction

   (b) A floor and roof with drainage control. If a floor drain is provided, a check valve or its equal is included

   (c) Doors and frames designed to comply with the requirements of a minimum 2-hour fire rating

For further discussion of missile protection, see Section 3.5.

1.8.1.116 Conformance to Regulatory Guide 1.116, Revision 0-R, June 1976: Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

HCGS complies with the quality assurance requirements of ANSI N45.2.8-1975, as modified by Regulatory Guide 1.116, except as discussed in the following paragraphs:

In Section 1.1, Scope, ANSI 45.2.8-1975, the term "important items" is interpreted to apply to those activities or quality attributes of an item or service that could affect a nuclear safety-related characteristic. For example, if a barrier is required for leakage control but serves no structural function, the leak-tight characteristic is considered "important", but appearance, dimensional requirements, and structural features would not necessarily be considered important. If a pump casing is required for coolant boundary integrity, but the pump does not have to operate to provide for nuclear safety, those attributes that affect its operation would not be considered important from the standpoint of nuclear safety.

Contrary to Section 2.1, Planning, ANSI 45.2.8-1975, the required planning is frequently performed on a generic basis for application to many installations on one or more projects. This results in standard procedures or plans for installation, inspection, and testing, which meet the requirements of the standard. Individual plans for each item or system are not normally prepared unless the work operations are unique. However, standard procedures or plans are reviewed for applicability in each case. Installation plans or procedures are also limited in scope to those actions or activities that are essential to maintain or achieve required quality.

In Section 3.3, Process and Procedures, ANSI N45.2.8-1975, the terms "installation site" and "site" are interpreted to mean the same as "construction site" (or station site during operation). When applied to documents, these may be at the central office or work area document control station. The term "installation area" is interpreted to mean the immediate proximity of the location where work is to be performed. CN 30

Section 3.5(e), Site Conditions, ANSI N45.2.8-1975, is applied only if subsequent correction of adjacent nonconformances would damage the item being installed.

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Contrary to Section 4.6, Care of Item, ANSI N45.2.8-1975, the constructor or construction manager (or licensee during operation) is assumed to be the "responsible organization" for temporary usage of equipment or facilities, unless specifically prohibited by contract or in writing from the owner. All other conditions and considerations for temporary use in Section 4.6 are applied.

See Chapter 17 for further discussion of quality assurance and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.117 <u>Conformance to Regulatory Guide 1.117, Revision 1, April 1978: Tornado Design Classification</u>

Although Regulatory Guide 1.117 is not applicable to HCGS, per its implementation section, HCGS complies with it.

For further discussion of tornado loadings, see Section 3.3.2.

1.8.1.118 <u>Conformance to Regulatory Guide 1.118, Revision 2, June 1978: Period Testing of Electric Power and Protection Systems</u>

Although Regulatory Guide 1.118 is not applicable to HCGS, per its implementation section, HCGS complies with it.

See Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.119 <u>Conformance to Regulatory Guide 1.119, Revision 0, June 1976: Surveillance Program for New Fuel Assembly Designs</u>

Regulatory Guide 1.119 was withdrawn by the NRC on June 23, 1977 and is not applicable.

not be located inside buildings containing safety-related equipment. If above-ground tanks are used, they should be located at least 50 feet from any building containing safety-related equipment or, if located within 50 feet, they should be housed in a separate building with construction having a minimum fire resistance rating of 3 hours. Potential oil spills should be confined or directed away from buildings containing safety-related equipment. Totally buried tanks are acceptable outside or under buildings. See NFPA 30, Flammable and Combustible Liquid Code, for additional guidance.

At HCGS, the two 26,500-gallon diesel fuel oil storage tanks are located in each of the storage tank rooms at floor elevation 54 feet of the auxiliary building diesel generator area. There are four of these rooms containing a total of 212,000 gallons of fuel oil. Each room is enclosed by 3-hour fire barriers. The diesel area is separated from the control area by 3-hour fire walls. A manually actuated deluge sprinkler system, a fixed automatic carbon dioxide total flooding system, and an automatic fire detection system is provided for each room The above diesel fuel oil storage meets the requirement of NFPA 30.

Although the combustible loading in the diesel fuel oil storage tank rooms is 7,045,000 Btu/ft^2 of floor area, oxygen depletion can restrict the fully developed period of any fire event to approximately 5 minutes in consideration of the postulated combustion of approximately 17.36 gallons of fuel oil.

HCGS's fuel oil storage conforms to the requirements of Appendix R to 10 CFR 50 and BTP CMEB 9.5.1, Revision 2.

1.8.1.121 Conformance to Regulatory Guide 1.121, Revision 0, August 1976: Bases for Plugging Degraded PWR Steam Generator Tubes

Regulatory Guide 1.121 is not applicable to HCGS.

1.8.1.122 Conformance to Regulatory Guide 1.122, Revision 1, February 1978: Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

Although Regulatory Guide 1.122 is not applicable to HCGS, per its implementation section, HCGS complies with it.

For further discussion of seismic design, see Sections 3.7 and 3.10.

1.8.1.123 Conformance of Regulatory Guide 1.123, Revision 1, July 1977: Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

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HCGS compiles with Regulatory Guide 1.123 ~~D~~during construction and startup phases, subject to clarifications stated below. During the operations phase, item a clarification applies only.

The architect-engineer indicates that the original HCGS project commitment was to ANSI N45.2.13 (Draft October 1973) rather than to ANSI N45.2.13-1976. The architect-engineer NQAM has been revised to reference the 1976 document, as modified by the Regulatory Guide, subject to the following specific changes:

a. Regulatory Guide Section C.2 - This section requires the application of elements of the ASME B&PV Code, Section III, Divisions 1 and 2, and Section XI; and ANSI N45.2.13-1976; specifically, those elements not covered by the ASME B&PV Code for procurement of ASME B&PV Code items and services. The architect-engineer takes exception to the requirement, and has the following alternate position:

   The application of the ASME B&PV Code requirements above to the procurement of ASME B&PV Code items and services is adequate, based on the fact that ASME B&PV Code represents the composite knowledge and experience of a large segment of the nuclear industry, that the ASME B&PV Code is constantly being reevaluted for adequacy, that addenda are issued frequently, and that, to our knowledge, historical data do not exist that would indicate that the ASME B&PV Code quality assurance requirements, relative to the procurement of ASME B&PV items and services, are inadequate.

b. Regulatory Guide Section C.2 - This section of the regulatory position appears to be inconsistent. It states that the purchase should verify the implementation of the suppliers corrective action system "....when such a system is required." This implies that it is not always necessary to require a supplier to have a corrective action system. The next sentence, however states, "While Section 9.0 of ANSI N45.2.13-1976 addresses elements of the purchaser's corrective action system, these same elements are applicable to the supplier's corrective action system." This sentence could be interpreted as always requiring that a corrective action system be part of the supplier's quality assurance program. The architect-engineer's interpretation of this section is as follows:

A corrective action system may, depending upon complexity and/or importance to safety of the item or service procured, be imposed upon the supplier. When a corrective action system is imposed upon a supplier, the applicable elements of ANSI Section 9.0 are included, and its implementation is verified by the architect-engineer.

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c. Regulatory Guide Section C.4. This section requires that the portion of a procurement document identifying the method of acceptance of an item or service be on hand during receipt inspection for use by receiving inspection personnel. The architect-engineer's interpretation of this requirement is as follows:

The applicable portions of procurement documents are available at the location, shop, jobsite, etc, where an inspection activity is being performed. The architect-engineer's quality program uses source surveillance inspection performed by the procurement inspection department to ensure that supplied items meet procurement document requirements.

The architect-engineer receiving personnel use a release document from procurement inspection department personnel to assist in determining acceptance of an item. Those procurement documents that must be available to receiving inspection personnel are applicable only to inspections/surveillance being performed at the time of receipt.

d. Regulatory Guide Section C.6.a. The above reference requires that guidelines contained in ANSI N45.2.13-1976, Section 4.2a, indicated by the verb "should", be considered of sufficient safety importance to be treated in the same manner as requirements of the ANSI Standard. The architect-engineer's position relative to ANSI N45.2-13-1976, Section 4.2a, concerning evaluation of supplier sources based on historical supplier data, is as follows:

When evaluation of a supplier is based solely on historical supplier data, these data include, primarily, architect-engineer's records accumulated in connection with previous procurement actions. Data documenting the operating experience of identical or similar products of the prospective supplier are used when they become available to architect-engineer; however, such data is under the control of other firms and utilities.

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e. Regulatory Guide Section C.6.c. The above reference requires that guidelines contained in ANSI N45.2.13-1976, Section 10.2, indicated by the verb "should", be considered of sufficient safety importance to be treated in the same manner as requirements of the standard.

f. The architect-engineer's position relative to ANSI N45.2.13-1976, Section 10.2.d, concerning the person attesting to the certificate, is as follows:

The person attesting to a certificate for a supplier must be identified as an authorized and responsible employee of the supplier, in supplier's quality assurance manual, or by letter from the supplier.

g. The architect-engineer's position relative to ANSI N45.2.13-1976, Section 10.2.f, concerning the verification of the validity of supplier certificates and the effectiveness of the certification system, is as follows:

The verification of the validity of supplier certificates and the effectiveness of the certification system functions as an integral part of the total supplier control and product acceptance program, and no separate architect-engineer system exists that addresses itself solely to such verification. The