NUREG-1048
Supplement No. 1

Safety Evaluation Report
related to the operation of Hope Creek Generating Station

Docket No. 50-354

Public Service Electric and Gas Company
Atlantic City Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1985

NOTICE

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Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

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Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $4.50

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ABSTRACT

Supplement No. 1 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In October 1984, the U.S. Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-1048) on the application filed by Public Service Electric and Gas Company (applicant), acting on behalf of itself and Atlantic City Electric Company, for a license to operate the Hope Creek Generating Station (Docket No. 50-354). At that time, the staff identified items that were not yet resolved with the applicant. The purpose of this supplement to the SER is to provide the staff evaluation of open items that have been resolved, to report on the status of all open items, and to address those recommendations that are contained in the Advisory Committee on Reactor Safeguards (ACRS) letter of December 18, 1984.

During its 296th meeting on December 13-15, 1984, the ACRS reviewed the operating license application filed by the applicant. The Committee, in a December 18, 1984, letter from Chairman Jesse C. Ebersole to NRC Chairman Nunzio J. Palladino, concluded that subject to the resolution of open items identified by the staff in the SER and the items noted in the above referenced letter and satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that Hope Creek can be operated at power levels up to 3,293 megawatts-thermal (100% power) without undue risk to the health and safety of the public. The ACRS letter is presented in Appendix H of this supplement.

Each of the following sections or appendices of this SER supplement is numbered the same as the corresponding SER section or appendix that is being updated. Appendix A is a continuation of the chronology of the staff's actions related to the processing of the Hope Creek application. Appendix B is a list of references cited in this report.* Appendix D is a list of acronyms used herein. Appendix E identifies principal contributors to this SER supplement.

Appendix G is a revised report addressing the control of heavy loads at Hope Creek. Appendix H presents a copy of the letter from the ACRS to the NRC Chairman regarding Hope Creek. Appendix I presents the Technical Evaluation Report of the Detailed Control Room Design Review prepared by the Lawrence Livermore National Laboratory. Appendix J contains errata to the SER.

Copies of this SER supplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the Pennsville Public Library, 190 South Broadway, Pennsville, New Jersey. They are also avilable for purchase from the sources indicated on the inside front cover of this report.

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*Availability of all material cited is described on the inside front cover of this report.

The NRC Project Manager assigned to the operating license application for Hope Creek is Mr. David H. Wagner. Mr. Wagner may be contacted by writing to

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1.7 Outstanding Issues

The staff identified certain outstanding issues in the SER that had not been resolved with the applicant. The status of these issues is listed in an updated version of Table 1.2 and discussed further in the indicated sections of this report. If the staff review is completed for an issue, it is indicated as "closed." The staff will complete its review of outstanding issues before the operating license is issued.

1.8 Confirmatory Issues

The staff identified confirmatory items in the SER that required additional information to confirm preliminary conclusions. The status of these items is listed in an updated version of Table 1.3 and discussed further in the indicated sections of this report. If the staff review is completed for an item, it is identified as "closed."

1.11 Nuclear Waste Policy Act of 1982

Section 302(b) of the Nuclear Waste Policy Act of 1982 states that NRC shall not issue or renew a license for a nuclear power reactor unless the utility has signed a contract with the Department of Energy for disposal services. Public Service Electric & Gas Company has signed a contractual agreement with the Department of Energy dated June 13, 1983.

Table 1.2 Outstanding issues

Issue	Status	SER section(s)
(1) Riverborne missiles	Awaiting information	
(2) Equipment qualification	Awaiting information	
(3) Preservice inspection program	Awaiting information	
(4) GDC 51 compliance	Awaiting information	
(5) Solid-state logic modules	Under review	
(6) Postaccident monitoring instrumentation	Under review	
(7) Minimum separation between non-Class 1E conduit and Class 1E cable trays	Awaiting information	
(8) Control of heavy loads	Closed	9.1.5
(9) Alternate and safe shutdown	Under review	
(10) Delivery of diesel generator fuel oil and lube oil	Closed	9.5.4.2
(11) Filling of key management positions	Awaiting information	
(12) Training program items		
(a) Initial training programs	Under review	
(b) Requalification training programs	Under review	
(c) Replacement training programs	Under review	
(d) TMI issues I.A.2.1, I.A.3.1, and II.B.4	Under review	
(e) Nonlicensed training programs	Under review	
(13) Emergency dose assessment computer model	Under review	
(14) Procedures generation package	Under review	
(15) Human factors engineering	Awaiting information	

Table 1.3 Confirmatory issues

Issue	Status	SER section(s)
(1) Feedwater isolation check valve analysis	Awaiting information	
(2) Plant-unique analysis report	Awaiting information	
(3) Inservice testing of pumps and valves	Awaiting information	
(4) Fuel assembly accelerations	Awaiting information	
(5) Fuel assembly liftoff	Awaiting information	
(6) Review of stress report	Awaiting information	
(7) Use of Code cases	Awaiting information	
(8) Reactor vessel studs and fasteners	Awaiting information	
(9) Containment depressurization analysis	Under review	
(10) Reactor pressure vessel shield annulus analysis	Under review	
(11) Drywell head region pressure response analysis	Under review	
(12) Drywell-to-wetwell vacuum breaker loads	Awaiting information	
(13) Short-term feedwater system analysis	Awaiting information	
(14) Loss-of-coolant-accident analysis	Awaiting information	
(15) Balance-of-plant testability analysis	Awaiting information	
(16) Instrumentation setpoints	Awaiting information	
(17) Isolation devices	Awaiting information	
(18) Regulatory Guide 1.75	Under review	
(19) Reactor mode switch	Under review	
(20) Engineered safety features reset controls	Awaiting information	

Table 1.3 (Continued)

Issue	Status	SER section(s)
(21) High pressure coolant injection initiation	Awaiting information	
(22) IE Bulletin 79-27	Awaiting information	
(23) Bypassed and inoperable status indication	Awaiting information	
(24) Logic for high pressure coolant injection interlock circuitry	Awaiting information	
(25) End-of-cycle recirculation pump trip	Awaiting information	
(26) Multiple control system failures	Under review	
(27) Relief function of safety/relief valves	Under review	
(28) Main steam tunnel flooding analysis	Awaiting information	
(29) Cable tray separation testing	Awaiting information	
(30) Use of inverter as isolation device	Awaiting information	
(31) Core damage estimate procedure	Awaiting information	
(32) Continuous airborne particulate monitors	Awaiting information	
(33) Qualifications of senior radiation protection engineer	Awaiting information	
(34) Onsite instrument information	Awaiting information	
(35) Airborne iodine concentration instruments	Awaiting information	
(36) Emergency Plan items	Awaiting information	
(37) TMI Item II.K.3.18	Awaiting information	

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.5 Overhead Heavy Load Handling Systems

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was developed. Following the issuance of NUREG-0612, a generic letter dated December 22, 1980, was sent to all operating plants, applicants for operating licenses, and holders of contruction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. As indicated above, in accordance with the generic letter dated December 22, 1980, the Public Service Electric and Gas Company, the applicant for Hope Creek Generating Station, was requested to review his provisions for the handling and control of heavy loads at the Hope Creek facility to determine the extent to which the guidelines of NUREG-0612 are satisfied and to commit to mutually agreeable changes and modifications that would be required to fully satisfy these guidelines.

In the SER, the staff could not conclude that the Hope Creek heavy load handling systems were in compliance with the criteria contained in NUREG-0612. The applicant, by letters dated June 28, 1983, and September 7, 1984, provided responses to the guidelines of NUREG-0612; however, these responses were incomplete. An additional submittal dated November 5, 1984, provided sufficient detail for the staff to complete its review.

The staff and its consultant, the Idaho National Engineering Laboratory (INEL), have reviewed the applicant's submittals for Hope Creek. As a result of its review, INEL has issued a technical evaluation report (TER). The staff has reviewed the TER and concurs with its findings that the guidelines in NUREG-0612, Section 5.1.1, have been satisfied. The TER is included as Appendix G to this supplement. The staff concludes that the guidelines of NUREG-0612 for Hope Creek are satisfied and that no further action is required concerning Sections 5.1.2 through 5.1.5 of NUREG-0612.

Additionally in the SER the staff stated that the applicant had not provided information concerning the potential effects due to the failure of the concrete plugs while being handled by the reactor building crane. By submittal dated January 18, 1985, the applicant stated that the refueling channel slot plugs and dryer-separator pool plugs are reinforced concrete encased in steel. The reactor cavity shield plugs are steel-reinforced concrete with the edges enclosed in a steel channel. Therefore, the staff concludes that breaking of the concrete for these plugs into pieces is not credible and the Hope Creek design is acceptable.

9.5 Other Auxiliary Systems

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System

In the SER, the staff had a concern regarding how the applicant could refill the diesel engine fuel oil and lube oil storage tanks during site flooding conditions. The applicant in a letter dated November 2, 1984, provided additional information to address the staff concern. First, the applicant stated that it is unlikely that the site will flood above plant grade elevation. This is based on the fact that the highest historical high water level recorded at this site, which occurred in November 1950, was 4 ft below established plant grade.

Second, the applicant stated that under probable maximum flood (PMF) conditions (which would result in the most severe site flooding), it would be unrealistic to expect site flooding to persist for more than 24 hours. However, should flood conditions persist even for a substantially longer period of time, the staff believes there would be adequate time to arrange for delivery of fuel or lube oil either by truck or barge. Getty, Texaco, and the Sun Oil Company have refineries within a 75-mi radius of the site to ensure fuel or lube oil delivery to the site within the required period. Similarly, the applicant's commitment to refuel within a 7-day period provides ample time to clear roads of any credible snowfall or debris that may accumulate as the result of the storm.

And third, during an actual shutdown as the result of a PMF with loss of offsite power, all four diesel generators would not be required to achieve and maintain cold shutdown. On the basis of a time-dependence generator loading, there would in fact be approximately 14 days of fuel supply available for required diesel generator operation. On this basis there would be sufficient time to refuel the storage tanks under any credible occurrence.

The staff has reviewed the applicant's submittal and concurs that sufficient fuel and lube oil will be stored on site to power the diesel generators on a time-dependence loading for a period of 14 days and in the event of a PMF with loss of offsite power. The staff concurs with the applicant that within this time period, there should be ample time to take appropriate actions necessary to permit delivery of fuel and lube oil to the site. The staff finds this acceptable.

As a further precaution, the applicant has committed to implement procedures that will require filling of all diesel generator fuel storage tanks to capacity in the event of an alert of an impending hurricane, tornado, or tropical storm.

18 HUMAN FACTORS ENGINEERING

18.1 Detailed Control Room Design Review

Item I.D.1, "Control Room Design Reviews," of Task I.D, "Control Room Design," of the NRC Action Plan (NUREG-0660) developed as a result of the accident at Three Mile Island (TMI), Unit 2, states that licensees and applicants for operating licenses will be required to perform a detailed control room design review (DCRDR) to identify and correct design discrepancies. The objective, as stated in NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent or cope with accidents if they occur by improving the information provided to them. Supplement 1 to NUREG-0737 confirmed and clarified the DCRDR requirement in NUREG-0660. As a result of Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct a DCRDR on a schedule negotiated with NRC.

NUREG-0700 describes four phases of the DCRDR to be performed by the applicant or licensee. The phases are

(1) planning
(2) review
(3) assessment and implementation
(4) reporting

Criteria for evaluating each phase are contained in Section 18.1 and Appendix A to Section 18.1 of the Standard Review Plan (NUREG-0800).

Supplement 1 to NUREG-0737 requires each applicant or licensee to submit a program plan that describes how the following elements of the DCRDR will be accomplished:

(1) establishment of a qualified multidisciplinary review team

(2) function and task analysis to identify control room operator tasks and information and control requirements during emergency operations

(3) a comparison of display and control requirements with a control room inventory

(4) a control room survey to identify deviations from accepted human factors principles

(5) assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected

(6) selection of design improvements

(7) verification that selected design improvements will provide the necessary correction

(8) verification that improvements will not introduce new HEDs

(9) coordination of control room improvements with changes from other programs such as the safety parameter display system, operator training, Regulatory Guide 1.97 instrumentation, and upgrade of emergency operating procedures

Supplement 1 to NUREG-0737 also requires each applicant or licensee to submit a summary report at the end of the DCRDR. The report should describe the proposed control room changes and implementation schedules and provide justification for leaving safety-significant HEDs uncorrected or partially corrected.

The NRC staff will evaluate the organization, process, and results of each DCRDR. The evaluation of the DCRDR effort will consist of the following, as described in NUREG-0800:

(1) an evaluation of the program plan report submitted by the applicant or licensee

(2) a visit to some of the plant sites to audit the progress of the DCRDR programs

(3) an evaluation of the DCRDR summary report submitted by the applicant or licensee

(4) a possible preimplementation audit

(5) the preparation of an SER that will present the results of the NRC staff's evaluation

Significant HEDs should be corrected. Improvements that can be accomplished with an enhancement program should be done promptly.

As required by Supplement 1 to NUREG-0737, the applicant will complete the DCRDR before licensing. By letter dated October 17, 1983, the applicant submitted the DCRDR program plan for Hope Creek. A meeting was held in Bethesda, Maryland, on February 2, 1984, to discuss the DCRDR. The staff formally provided its comments on the DCRDR program plan through the resultant meeting summary dated February 24, 1984. By letter dated April 10, 1984, the applicant submitted a description of the DCRDR function and task analysis methodology. The DCRDR Summary Report was submitted by letter dated August 14, 1984.

A human factors engineering preimplementation audit of the DCRDR was performed at the Hope Creek site on November 13 through 15, 1984, by an NRC team assisted by consultants from Lawrence Livermore National Laboratory (LLNL). Following the onsite audit, the applicant submitted a letter dated December 6, 1984, documenting commitments made during the preimplementation audit.

Consultants from LLNL have prepared a technical evaluation report, which is included as Appendix I to this supplement. The NRC staff concurs with the technical evaluations, conclusions, and recommendations contained in the LLNL report.

The staff concludes that the applicant will satisfy the requirements of Supplement 1 to NUREG-0737 for a DCRDR of the Hope Creek Generating Station with the efforts completed to date and those in progress. To satisfactorily complete

the DCRDR required by Supplement 1 to NUREG-0737, the applicant must fulfill the commitments documented in the letter of December 6, 1984. The staff will prepare a supplement to the SER based on its review of the supplemental DCRDR Summary Report, which the applicant has committed to submit to the NRC by November 1985. The NRC staff might audit the implementation of the control room design modifications.

18.2 Safety Parameter Display System

The applicant will submit the safety analysis of the Hope Creek safety parameter display system in March 1985.

18.3 TMI Item II.K.3.27 - Common Reference Level

TMI Task Action Plan Item II.K.3.27 requires that a common reference level be provided for vessel level instrumentation. In Amendment 8 to the FSAR dated October 1984, the applicant states that a common reference point for instruments measuring water level in the reactor vessel will be established at Hope Creek. The establishment of a common reference point satisfies the requirement of TMI Task Action Plan Item II.K.3.27.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

A subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) considered the application for an operating license for the Hope Creek Generating Station at a meeting in Philadelphia, Pennsylvania, on November 28 and 29, 1984. The subcommittee visited the site and toured the facility on the morning of November 28, 1984. On the afternoon of November 28, 1984, and all day November 29, 1984, the ACRS subcommittee held a public meeting to discuss the Hope Creek application. The full committee reviewed the application at its 296th meeting on December 13-15, 1984. A copy of the Committee's report to NRC Chairman Palladino dated December 18, 1984, is included as Appendix H to this supplement.

The Committee noted two areas in which it had specific recommendations for the applicant. First, the Committee recommended that the applicant prepare a structured test program for evaluating overspeed protection of the turbine. This recommendation is based on the nonoptimum orientation of the turbine relative to vital facility components. Second, the Committee Chairman urged the applicant to consider the applicability of an ultimate plant protection system (UPPS) such as that referenced in the General Electric GESSAR II design. By letter dated December 21, 1984, the staff requested the applicant to respond to these concerns.

Similarly, the Committee noted two areas in which it had specific recommendations for the staff. First, the Committee Chairman also urged the staff to consider the applicability of a UPPS at Hope Creek. Second, the Committee recommended that the staff give additional attention to the habitability requirements of the control room, including evaluations of the potential loss of both trains of the emergency ventilation system and the heat load and rate of temperature rise in the room under a range of heating, ventilation, and air conditioning conditions. The staff is reviewing the above recommendations and will provide responses in future SER supplements.

During the meeting, the NRC staff identified a number of open issues that must be resolved before the granting of an operating license. The ACRS letter notes that subject to the resolution of open items identified by the NRC staff and the items noted above and the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Hope Creek Generating Station can be operated at power levels up to 3,293 MWt without undue risk to the health and safety of the public.

APPENDIX A

CONTINUATION OF CHRONOLOGY

October 18, 1984	Letter from applicant forwarding revised FSAR Question 430.81 regarding protective coatings of interior and exterior surfaces of fuel oil storage tank against corrosion.
October 19, 1984	Letter from applicant forwarding review of updated master listing of seismic and dynamic qualification summary and status of safety-related equipment for January 1985 audits.
October 23, 1984	Letter to applicant forwarding request for additional information on preservice program for review of application for operating license (OL).
October 24, 1984	Letter to applicant concerning November 13-15, 1984, preimplementation audit of detailed control room design review to evaluate results of review to date.
October 25, 1984	Letter from applicant forwarding updated system-sorted master listing of seismic and dynamic qualification summary and status of safety-related equipment.
October 29, 1984	Letter from applicant forwarding tabulation of FSAR commitments through September 1984 and their corresponding resolutions.
November 2, 1984	Letter from applicant forwarding revised responses to FSAR Questions 410.69 and 430.88 and summary of reduced standby diesel generator loading to accommodate standing hurricane after loss of offsite power.
November 2, 1984	Letter from applicant forwarding Amendment 8 to FSAR.
November 5, 1984	Letter from applicant forwarding revisions to FSAR Table 9.1-10 and Section 9.1.5.6, per October 29, 1984, discussions on response to heavy loads - Phase I.
November 9, 1984	Letter from applicant forwarding Revision 5 to Emergency Plan.
November 13, 1984	Letter from applicant forwarding affidavit certifying distribution of Amendment 8 to FSAR, per 10 CFR 2.101.

November 16, 1984	Letter from applicant forwarding addendum to updated master equipment list for seismic and dynamic qualification summary and status of safety-related equipment and revised active pump list.
November 16, 1984	Letter to applicant forwarding request for additional information on applicant's February 10, 1984, plant-unique analysis report.
November 21, 1984	Letter to applicant forwarding request for additional information on equipment qualification for review of OL application, including information on TMI Items II.E.4.2 and II.D.1 regarding purge and vent valve operability and testing of BWR safety/relief valves, respectively.
November 28, 1984	Summary of November 20, 1984, meeting with applicant in Bethesda, Maryland, concerning training program.
December 5, 1984	Letter to applicant requesting notification within 60 days of plans for ensuring that plant is designed in accordance with FSAR commitment.
December 6, 1984	Letter from applicant discussing preimplementation audit commitments regarding control room design.
December 11, 1984	Letter to applicant requesting that equipment qualification program be expedited per October 19, 1984, letter and NRC be provided with updated summary when 85% completion level is reached.
December 12, 1984	Letter to applicant forwarding comments on protection of safety-related structures, systems, and components from missiles generated by natural phenomena.
December 17, 1984	Letter from applicant forwarding revised response to Generic Letter 83-28 concerning post-trip review system.
December 17, 1984	Letter from applicant informing of intended issuance of Amendment 9 to FSAR by February 15, 1985, enclosing table of FSAR commitments for October and November 1984, their corresponding resolutions, and response to commitment to be included in FSAR Amendment 9.
December 18, 1984	Summary of November 26, 1984, meeting with applicant and EG&G in Bethesda, Maryland, concerning preservice inspection program.
December 21, 1984	Letter to applicant forwarding Advisory Committee on Reactor Safeguards (ACRS) report and requesting applicant's position on ACRS recommendations concerning turbine overspeed protection test program and applicability of ultimate plant protection system.

December 27, 1984	Letter to applicant concerning certification of compliance to 10 CFR 50.49. (Generic Letter 84-24)
December 28, 1984	Letter from applicant forwarding Revision 1 to Training Program TP-305C, "NRC Licensed Operator Requalification Program," per SER Open Item 12(b).
December 28, 1984	Letter to applicant discussing review of FSAR regarding compliance with 10 CFR 50, Appendix G.
January 7, 1985	Letter to applicant forwarding marked-up pages to FSAR addressing SER Open Item 12 concerning training programs and description of emergency dose assessment computer model in response to SER Open Item 13.
January 8, 1985	Letter from applicant responding to November 16, 1984, request for additional information on plant-unique analysis report.
January 9, 1985	Letter to applicant concerning Fire Protection Policy Steering Committee report. (Generic Letter 85-01)
January 9, 1985	Letter to applicant requesting information for antitrust review of OL application concerning any changes since March 1, 1983, in response to Regulatory Guide 9.3.
January 9, 1985	Letter to applicant forwarding request for additional information on review of riverborne missiles as discussed at December 18, 1984, meeting.
January 15, 1985	Letter from applicant forwarding commitment status of responses to NRC requests for additional information on FSAR and amendments through December 1984.
January 16, 1985	Letter from applicant forwarding Revision 6 to Emergency Plan.
January 16, 1985	Letter to applicant forwarding approved December 20, 1984, notice of environmental assessment and finding of no significant impact concerning consideration of granting relief from 10 CFR 50.55a requirements for reactor coolant pressure boundary equipment and Quality Group A components.
January 17, 1985	Letter from applicant forwarding Revision 0 to draft Technical Specifications, based on BWR-4 Standard Technical Specifications.
January 18, 1985	Letter from applicant forwarding supplementary response to SER Open Item 8 concerning effects of failure of shield plug while it was being removed before refueling.
January 23, 1985	Letter to applicant forwarding excerpts from SER, identifying additional information needed to complete review.

January 25, 1985	Letter to applicant forwarding request for additional information on TMI Action Plan Item II.K.3.28, "Qualification of Accumulators on Automatic Depressurization System Valves."
January 28, 1985	Letter from applicant forwarding procedures generation package, to be incorporated into Amendment 10 of FSAR and emergency operating procedures writers guide, closing out SER Open Item 14.
January 31, 1985	Letter from applicant forwarding response to staff's November 16, 1984, request for additional information on plant-unique analysis report needed to complete review of hydrodynamic loads.
February 1, 1985	Letter from applicant requesting permission to adopt the provisions of NB-2510 of the 1983 Edition, Summer 1983 Addenda of the ASME Code.
February 1, 1985	Letter from applicant forwarding responses to staff's letter of November 21, 1984, concerning equipment qualification.
February 2, 1985	Letter from applicant forwarding SER response schedule per staff's request of January 23, 1985.
February 7, 1985	Letter from applicant forwarding Revision 1 to the Hope Creek Technical Specifications.
February 9, 1985	Letter from applicant forwarding Amendment 9 to the Hope Creek Final Safety Analysis Report.
February 11, 1985	Letter to applicant advising that his August 20, 1984, response to Draft SER Open Item 103 is acceptable, with the exception of certain items.

APPENDIX B

BIBLIOGRAPHY

U.S. Nuclear Regulatory Commission, Generic Letter from D. G. Eisenhut to all licensees of operating plants and applicants for operating licenses and holders of contruction permits, "Control of Heavy Loads," Dec. 22, 1980.

---, NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," Vol. 1, May 1980; Rev. 1, Aug. 1980.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0700, "Guidelines for Control Room Design Reviews," Sept. 1981.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Supp. 1, Dec. 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

---, NUREG-1048, "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station," Oct. 1984.

---, Office of Inspection and Enforcement, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," Nov. 30, 1979.

APPENDIX D

ACRONYMS AND INITIALISMS

ACRS	Advisory Committee on Reactor Safeguards
DCRDR	detailed control room design review
GDC	general design criterion(a)
HED	human engineering discrepancy
IE	Office of Inspection and Enforcement
INEL	Idaho National Engineering Laboratory
LLNL	Lawrence Livermore National Laboratory
NRC	U.S. Nuclear Regulatory Commission
OL	operating license
PMF	probable maximum flood
SER	Safety Evaluation Report
TER	technical evaluation report
TMI	Three Mile Island
UPPS	ultimate plant protection system

APPENDIX E

PRINCIPAL STAFF CONTRIBUTORS AND CONSULTANTS

This supplement to the Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

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APPENDIX G

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS - HOPE CREEK GENERATING STATION

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
HOPE CREEK GENERATING STATION (HCGS), UNIT 1
(PHASE I)
Docket No. 50-354

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ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of compliancy with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for Hope Creek Generating Station, Unit 1.

EXECUTIVE SUMMARY

Hope Creek Generating Station Unit 1 is consistent with NUREG 0612, Article 5.1.1.

CONTENTS

TABLES

Control of Heavy Loads at Nuclear Power Plants

Hope Creek Generating Station, Unit 1

(Phase I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load handling policy and procedures at Hope Creek Generating Station, Unit 1. This evaluation was performed with the objective of assessing conformance to the general load handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff applicant criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to Public Service Electric and Gas Company, the applicant for Hope Creek Generating Station (HCGS) requesting that the applicant review provisions for handling and control of heavy loads at HCGS, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On June 24, 1983, Public Service Electric and Gas Company provided the initial response [4] to this request, a revised updated report September 7, 1984 [9], and a final resolution of discrepancies in a response of November 5, 1984 [10].

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize Public Service Electric and Gas Company's review of heavy load handling at HCGS, Unit 1 accompanied by EG&G's evaluation, conclusions, and recommendations. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 1200 lbs.

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

A current listing of overhead handling systems is provided in Table 2.1. This list reflects adjustments and additions made during the past year. A second Table 2.2 lists 45 hoists in 29 handling systems that have been examined and

TABLE 2.1. NONEXEMPT HEAVY LOAD HANDLING SYSTEMS HOPE CREEK UNIT 1

Item Number	Handling System	Tag Number	Capacity Tons	Building	Floor Elevation Feet	Design Standard
1	Polar Crane	10H200	150/10	Reactor	201	a, b
2	Personnel Air Lock Hoist	10H217	30	Reactor	102	c, d
3	Recirculation Pump Motor Hoist	1AH201	24	Reactor	102	c, d
		1BH201	24	Reactor	drywell	c, d
4	HPCI Pump and Turbine Hoist	1AH211	4	Reactor	54	c, d
		1BH211	4	Reactor	54	c, d
5	Main Steam Tunnel Underhung Crane	10H214	2.5	Reactor	102	a, d, f
		10H223		Reactor	102	a, d, f
6	Inboard MSIV Hoist	10H219	2	Reactor	102	d
7	Diesel Generator Underhung Crane	0AH301, 0BH301 0CH301, 0DH301	2	Control and Diesel Generator	102	c, d, e
8	Intake Structure Gantry Crane	00H500	30/15	Intake Structure	122/128	b, d, e
9	CRD Service Hoist	borrowed	~1	Reactor	102	a, b, d
10	SACS Pumps Hoist	borrowed	~3.1	Reactor	102	a, b, d
11	SACS Heat Exchanger Hoists	borrowed	<5	Reactor	102	a, b, d
12	Personnel Lock Shield Removal Hoist	1AH218	15	Reactor	102	b, c, d
		1BH218	15	Reactor	102	b, c, d

Design Standards

a. ANSI B30.2 Overhead and Gantry Cranes, Top Running Bridge, Multiple Girder.
b. CMAA 70 Electric Overhead Traveling Cranes.
c. HMI 100 Electric Wire Rope Hoists.
d. ANSI B30.16 Overhead Hoists, Underhung.
e. ANSI B30.11 Monorail Systems and Underhung Cranes.
f. ANSI B30.17 Overhead and Gantry Cranes, Top Running Bridge, Single Girder, Underhung Hoist.

TABLE 2.2. EXEMPT HEAVY LOAD HANDLING SYSTEMS HOPE CREEK GENERATING STATION UNIT 1

Handling System	Tag Number	Cap. Ton	Building	Floor Elevation Feet	Exemption Criteria*
Reactor Water Cleanup Filter/Demineralizer Hoist	1AH220 1BH220	10	Reactor	178-6	B
RCIC Pump and Turbine Hoist	10H212	3	Reactor	54	C
Turbine Building Bridge Crane	10H102	220/45	Turbine	137	B
Vacuum Breaker Valve Removal Hoist	10H207	2	Reactor	54	C
Main Stream Line Relief Valve Removal Hoist	10H202	1	Reactor	135.5	C
Feedwater Heater Removal Hoist	1AH103	24	Turbine	102	B
H&V Equipment Removal Hoist	10H104	15	Turbine	171	B
Motor Generator Set Hoist	0AH105	15	Turbine	137	B
Secondary Condensate Pump Hoist	10H106	15	Turbine	54	B
Reactor Feed Pump Hoist	1AH107 1BH107 1CH107	15 15 15	Turbine	137 137 137	B
Water Box Removal Hoist	10H109 10H110	12 12	Turbine	81 81	B B
Steam Packing Exhaust Hoist	10H115	10	Turbine	77	B
Turbine Generator Auxiliary Crane	00H100	10	Turbine	137	B
Steam Met Air Ejector Hoist	1AH117 1BH117 1CH117 10H117	8 8 8 8	Turbine	77	B B B
Water Box Removal Hoist	10H111 10H112	8 8	Turbine	81	B B
Chiller Tube Removal Hoist	10H118	5	Turbine	171	B
Emergency Air Compressor Hoist	10H114	4	Turbine	123	B
Main Air Compressor Hoist	00H113 10H113	3 3	Turbine	123	B
Vacuum Pump Water Cooler Hoist	10H116	2	Turbine	77	B

TABLE 2.2. (continued)

Handling System	Tag Number	Cap. Ton	Building	Floor Elevation Feet	Exemption Criteria*
Heating and Cooling Coil Removal Hoist	1AH119	1 1/2	Turbine	171	B
	1BH119	1 1/2			B
Solid Radwaste Monorail	00H316	1 1/2	Service and Radwaste	102	B
Solid Radwaste Bridge Crane	00H317	7 1/2	Service and Radwaste	135 1/2	B
Demineralizer Removal Hoist	00H302	10	Service and Radwaste	102	B
Decontamination Evaporator Hoist	00H305	7 1/2	Service and Radwaste	54	B
Equipment Decontamination Room Hoist	00H314	5	Service and Radwaste	102	B
Machine Shop Underhung Crane	0AH301	5	Service and Radwaste	102	B
	0BH301	5			B
	0CH301	5			B
	00H301	5			B
Waste Evaporator Recirculation Pump Hoist	00H309	2	Service and Radwaste	54	B
	00H310	2			B
Waste Evaporator Hoist	00H312	1	Service and Radwaste	87	B
	00H313	1			
Recombiner System Hoist	00H318	1 1/2	Control and Diesel Gen.	67 1/4	C
	00H318	2 1/2			C

*

A. Crane is located in a building or structure that contains no safety related or safe shutdown equipment.

B. The crane's load path does not pass over any safety related or safe shutdown equipment on the floor below or next lower elevation.

C. Although the crane's capacity is greater than 1200 lb, its dedicated load is lighter than 1200 lb.

found to meet acceptable criteria to be considered exempt from the consideration of NUREG 0612. The exemption criteria are listed in the table footnotes.

B. EG&G Evaluation

Information from the submittals indicate that a detailed assessment has been made. Table 2.1 has been expanded to include items Nos. 9, 10, and 11 that were not in the original submittals. The November 5, 1984 letter advises that these handling systems will meet the design requirements of Guideline 7, even though they are borrowed when load handling is required. Also, the letter advises that the SACS Heat Exchanger hoist loads will be components weighing less than 5 tons each. This clears confusion arising from references on pages 143-144 of the FSAR and Table 9.1.2 where the total load is indicated to be 9.2 tons.

The data of Table 2.2 provides suitable listing of and justification for exempt hoisting systems.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that the applicant has included all applicable hoists and cranes in their list of handling systems which must comply with the requirements of the general guidelines of NUREG-0612.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612,

Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1--Safe Load Paths
- Guideline 2--Load-Handling Procedures
- Guideline 3--Crane Operator Training
- Guideline 4--Special Lifting Devices
- Guideline 5--Lifting Devices (not specially designed)
- Guideline 6--Cranes (Inspection, Testing, and Maintenance)
- Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3.1 Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

> "Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members,

beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

The overhead load handling systems are designed to minimize the potential for heavy load drops on spent fuel pool or safe shutdown equipment by moving the loads in safe load paths to the extent practical. The load paths are not painted on the floor. They are omitted to avoid possible operator confusion in areas such as the refueling floor where multiple paths would cross.

The load paths are defined in the specific load handling procedures and shown on the drawings that are incorporated in the procedures. Deviations from the defined load paths require written alternative procedures approved by the plant safety review committee.

Because opaque plastic sheets may be taped to the floor where the potential for radioactive contamination exists, polar crane load paths painted on the refueling floor (elevation 201 ft) may not be visible. The alternative method that is used at HCGS for the polar crane is to make a person other than the crane operator (i.e., a signalman) responsible for assuring that the load path is followed. The signalman inspects the load path before the lift to ensure that it is clear, reviews the specific load handling procedure before the lift, and provides direction to the crane operator to ensure that the prescribed path is followed. The specific load handling procedures clearly

define the duties and responsibilities of the operator, the signalman, and any other members of the load handling party.

The appropriate polar crane load path is temporarily marked with rope or pylons to provide a visual reference for the operator. If it is not possible to temporarily mark the load path, permanent or temporary match marks are used to assist in positioning the bridge and/or trolley for the lift. The method of marking the load path is defined in each specific load handling procedure.

The reactor building polar crane is the only nonexempt cab-operated crane at HCGS. Other nonexempt cranes, except for the main steam tunnel underhung crane, are simple hoists on monorails where the load path cannot vary. Most lifts are short lifts where movement is limited to one coordinate axis in addition to the vertical. As described in Section 9.1.5.2.2.f, each of the monorails for the main steam tunnel underhung crane is mounted on end trucks that provide the capability for load movement in both coordinate axes in addition to the vertical. For these nonexempt, noncab-operated hoists the specific load handling procedures define whether a signalman is used and whether the load path will be marked.

B. EG&G Evaluation

The Applicant's assessment of the overhead load handling systems with regard to the safe load paths for transporting heavy loads has been upgraded to show consistency with the intent of Guideline 1. The alternative of using a signalman and temporary load path markers is an acceptable method to accomplish the need.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with the intent of Guideline 1.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

The applicant stated under "Design Bases" (FSAR, Section 9.1.5.1, Item i) that the cranes will be operated in compliance with written procedures meeting NUREG 0612 Article 5.1.1(2) Guideline 2. HCGS considers this a commitment for compliance.

B. EG&G Evaluation

The applicant has committed to compliance with NUREG 0612, Article 5.1.1(2).

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with the intent of Guideline 2.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [6]."

A. Summary of Applicant's Statements

The applicant expressed the intention to follow this guideline as one of the "Design Bases" (FSAR Section 9.1.5.1, Item j). HCGS considers this a commitment to comply with the requirements.

B. EG&G Evaluation

The applicant has committed to train the crane operators in accordance with the criteria of Chapter 2-3, ANSI B30.2-1976.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with Guideline 3.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [7]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

A current reevaluation of the HCGS special lifting device needs reveals that the plant will use nine devices. Specific information concerning them is provided in a revised, retitled Table 9.1-14 of the FSAR. The changes include recalculation of the factors of safety and detailed comparison with the requirements of ANSI N14.6-1978.

As shown in revised Table 9.1-14, the stress design factors for all but two of the Hope Creek special lifting devices meet or exceed the values of three versus yield strength and five versus ultimate strength required by ANSI N14.6-1978. The design of the two Hope Creek lifting devices (dryer-separator sling and RPV service platform sling) that do not meet the safety factor criteria is the same as the design of the corresponding Washington Nuclear Plant No. 2 and Limerick Generating Station special lifting devices. Therefore, as discussed in the August 24, 1984, telephone call between the applicant and the NRC, no additional information regarding compliance with ANSI N14.6-1978 is provided.

B. EG&G Evaluation

Based on Public Service Electric and Gas Company statements that their special lifting devices are identical to components which have been reviewed elsewhere and found to be satisfactory, the HCGS special lifting devices are considered to meet the intent of Guideline 4.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with the intent of Guideline 4.

2.3.5 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [8]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

The heavy load handling slings are designed and used in accordance with ANSI B30.9-1971. A dynamic load factor will not be added to the static load when a sling is for use with a hoist that has a maximum hoisting speed equal to, or less than 30 ft/min.

B. EG&G Evaluation

HCGS has committed to comply with Guideline 5. The exception to including a dynamic load factor is permitted in Synopsis of Issues Associated With NUREG 0612.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with the intent of Guideline 5.

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified

inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

Inspection, testing, and maintenance programs will meet ANSI B30 2-1976, except when the crane use is less frequent than the specified test or inspection frequency, the test or inspection is done prior to crane use.

B. EG&G Evaluation

Based on the applicant's committment that the inspection, testing, and maintenance program for cranes will meet ANSI B30.2-1976, EG&G considers that HCGS is consistent with Guideline 6.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with Guideline 6.

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [9]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

The applicable criteria of CMAA-70 and ANSI B30.2-1976 are used as a design basis of the overhead heavy load handling systems (FSAR, Section 9.1.5.1, Item m).

B. EG&G Evaluation

The applicant has identified the standards that were applied for the design of each crane and hoist listed in the table for nonexempt hoisting systems. In many cases the primary plus acceptable alternates both were applied.

C. EG&G Conclusions and Recommendations

Hope Creek Generating Station is consistent with Guideline 7.

2.4 Interim Protection Measures

The NRC staff has established (NUREG-0612, Article 5.3) that six measures should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Article 5.1, is complete. Four of these six interim measures consist of general Guideline 1, Safe Load paths; Guideline 2, Load-Handling Procedures; Guideline 3, Crane Operator Training; and Guideline 6, Cranes (Inspection, Testing, and Maintenance). The two remaining interim measures cover the following criteria:

- Heavy load technical specifications
- Special review for heavy loads handled over the core.

Note: Since HCGS is not operational the Interim Protection Measures are not applicable.

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 is apparently complete (see Section 2.2.1).

3.2 Guideline Recommendations

Compliance with the seven NRC guidelines for heavy load handling (Section 2.3) are satisfied at Hope Creek Generating Station, Unit 1. This conclusion is represented in tabular form as Table 3.1.

Guideline	Recommendation
1. Section 2.3.1 Safe load Paths	a. HCGS is consistent with the intent of Guideline 1.
2. Section 2.3.2 Load Handling Procedures	a. HCGS is consistent with the intent of Guideline 2.
3. Section 2.3.3 Crane Operator Training	a. HCGS is consistent with Guideline 3.
4. Section 2.3.4 Special Lifting Devices	a. HCGS is consistent with the intent of Guideline 4.

TABLE 3.1. HOPE CREEK GENERATING STATION, UNIT 1/NUREG-0612 COMPLIANCE MATRIX

Equipment Designation	Heavy Loads	Capacity Tons	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline Crane Design
1. Reactor Building Polar Crane	23 items listed, 0.8 to 110 tons	150/10	C	C	C	C	C	C	C
2. Personnel Air Lock Hoist	3 times listed, 17 to 30 tons	30	C	C	C	--	C	C	C
3. Recirculation Pump Motor Hoist	Recirculation pump motor, 24 tons	24	C	C	C	--	C	C	C
4. HPCI Pump and Turbine Hoist	HPCI pump and turbine parts, 3.75 tons	4	C	C	C	--	C	C	C
5. Main Steam Tunnel Underhung Hoist	MSIV operators, 1.22 tons	2.5	C	C	C	--	C	C	C
6. Inboard MSIV Hoist	MSIV operators, 1.22 tons	2	C	C	C	--	C	C	C
7. Diesel Generator Underhung Hoist	Diesel-generator parts, 1.77 tons	2	C	C	C	--	C	C	C
8. Intake Structure Gantry Crane	Intake Structure parts, 19 tons	30/15	C	C	C	--	C	C	C
9. CRD Service Hoist	Neutron monitoring cask <1150 lb maintenance equipment 2000 lb	~1	C	C	C	--	C	C	C
10. SACS Pumps Hoist	Pump motors	~3.1	C	C	C	--	C	C	C
11. SACS Heat Exchanger Hoists	Heat exchanger return end cover 18400 lb	<5	C	C	C	--	C	C	C
12. Personnel Lock Shield Removal Hoist	T-shaped shield block, 21 tons	15	C	C	C	--	C	C	C

C = Applicant action consistent with NUREG-0612 guideline.

NC = Applicant action not consistent with NUREG-0612 guideline.

-- = Guideline is not applicable to this handling system.

I = Insufficient information was provided to determine consistency.

Guideline	Recommendation
5. Section 2.3.5 Lifting Devices Not Specially Designed	a. HCGS is consistent with the intent of Guideline 5.
6. Section 2.3.6 Cranes, (inspection testing and maintenance)	a. HCGS is consistent with Guideline 6.
7. Section 2.3.7 Crane Design	a. HCGS is consistent with Guideline 7.

3.3 Interim Protection

The actions for interim protection are not applicable to plants that are under construction.

3.4 Summary

HCGS is consistent with NUREG 0612, Article 5.1.1.

4. REFERENCES

1. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 17 May 1978.

3. USNRC, Letter to Public Service Electric and Gas Company. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 22 December 1980.

4. A. Singh (NRC), letter to EG&G Idaho, Inc. Subject: Public Service Electric and Gas Company's Article 9.1.5 of FSAR, "Overhead Heavy Load Handling System," July 28, 1983.

5. ANSI B30.2-1976, "Overhead and Gantry Cranes".

6. ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials".

7. ANSI B30.9-1971, "Slings".

8. CMAA-70, "Specifications for Electric Overhead Traveling Cranes".

9. A. Singh (NRC), letter to EG&G Idaho Inc., concerning "Control of Heavy Loads, Case Reviews" Transmitting revised FSAR Article 9.1.5 September 7, 1984.

10. Robert L. Mittl, Public Service Electric and Gas, letter to NRC, Subject: "Response to Heavy Loads--Phase I, Hope Creek Generating Station, Docket No. 50-354" dated November 5, 1984.

APPENDIX H

REPORT FROM THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 18, 1984

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE HOPE CREEK GENERATING STATION

During its 296th meeting, December 13-15, 1984, the Advisory Committee on Reactor Safeguards reviewed the application of Public Service Electric and Gas Company (the Applicant), acting on behalf of itself and as agent for the Atlantic City Electric Company, for a license to operate the Hope Creek Generating Station. The ACRS commented on the construction permit application for the Hope Creek Generating Station in a report dated February 28, 1974. Members and consultants of the Hope Creek Subcommittee toured the facility on November 28, 1984 and met in Philadelphia, Pennsylvania on November 28 and 29, 1984 to discuss the application. During our review, we had the benefit of discussions with representatives and consultants of the Applicant, General Electric Company, Bechtel Power Corporation, and the NRC Staff. We also had the benefit of the documents referenced.

The Hope Creek Generating Station consists of one unit and is immediately adjacent to the Salem Nuclear Generating Station. Both Stations are located on Artificial Island in Salem County, New Jersey, which is approximately 18 miles south of Wilmington, Delaware. The nearest densely populated center of 25,000 or more persons is Newark, Delaware, which is approximately 18 miles northwest of the Stations. Hope Creek uses a boiling water reactor (BWR/4) with a rated power level of 3293 MWt. The nuclear reactor is similar to other previously reviewed BWRs, such as the Limerick Generating Station, the Susquehanna Steam Electric Station, and the Edwin I. Hatch Nuclear Plant. The Hope Creek primary containment is a Mark I steel vessel and the secondary containment is reinforced concrete. The pressure suppression chamber is a torus shaped steel vessel which encircles the drywell at a lower elevation.

During our meeting, the NRC Staff identified a number of open issues that must be resolved prior to the granting of an operating license. We believe that these can be resolved in a manner satisfactory to the NRC Staff. We wish to be kept informed.

We heard a report from a representative of the NRC's Region I Office that the construction quality and quality assurance effectiveness at Hope Creek were satisfactory. He indicated that there is good communication at the site and that management attention is evident.

The liquefaction potential of the soils associated with plant-related structures was evaluated by the Applicant. The Applicant indicates that soils surrounding safety-related structures are stable against liquefaction at the design basis earthquake of 0.2g. The NRC Staff agrees that none of these soils will liquefy at levels up to the design basis earthquake. We agree with the NRC Staff.

Because of the nonoptimum orientation of the turbine relative to vital components in this plant, we recommend that a structured test program for evaluating overspeed protection of the turbine be prepared and submitted to the NRC Staff for review and approval before full power operation. We wish to be kept informed.

Although the control room at the Hope Creek Generating Station has been reviewed with respect to human factors, we encourage the NRC Staff to give additional attention to its habitability requirements. This should include evaluations of the potential loss of both trains of the emergency ventilation system and the heat load and rate of temperature rise in the room under a range of HVAC conditions.

We believe that, subject to the resolution of open items identified by the NRC Staff and the items noted above, and subject to the satisfactory completion of construction, staffing, and preoperational testing, there is reasonable assurance that the Hope Creek Generating Station can be operated at power levels up to 3293 MWt without undue risk to the health and safety of the public.

Additional comments by ACRS Member Jesse C. Ebersole are presented below.

Sincerely,

Jesse C. Ebersole

Jesse C. Ebersole
Chairman

Additional Comments by ACRS Member Jesse C. Ebersole

The Applicant has indicated that there will be an investigation of the current proposals by some BWR owners and by the General Electric Company to provide a simplified system to:

1. Provide an independent means to depressurize the primary coolant system.

2. Provide low pressure feedwater from a variety of sources using a small engine-driven pump or pumps.

3. Provide containment venting of steam after scrubbing through the suppression pool.

The minimum instrumentation for this system would be simple level indicators. The current GESSAR II design refers to this system as UPPS; however, the actual configuration of the system is still being considered.

The apparent overall simplicity and modest cost of this system and, if appropriately designed, the potential flexibility of the system to protect both core and containment cooling against a large number of accidents and system malfunctions would appear to justify careful consideration by both the Applicant and the NRC Staff as to its applicability to this plant.

References:

1. Public Service Electric and Gas Company, "Final Safety Analysis Report, Hope Creek Generating Station Unit 1," Volumes 1-20 and Amendments 1-8
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of Hope Creek Generating Station," USNRC Report NUREG-1048, dated October 1984
3. Letter dated November 23, 1984 from Richard W. Starostecki, NRC Region I to Chester Siess, ACRS, enclosing NRC Region I Evaluation of Construction Quality at Hope Creek Generating Station as of November 1984, Presented to ACRS Subcommittee November 28-29, 1984
4. Letter dated December 12, 1984 from Bruce A. Preston, Public Service Electric & Gas Co., to C. P. Siess, ACRS, attaching responses to questions from the ACRS Subcommittee meeting of November 28-29, 1984

APPENDIX I

DETAILED CONTROL ROOM DESIGN REVIEW

TECHNICAL EVALUATION REPORT
OF THE
DETAILED CONTROL ROOM DESIGN REVIEW
FOR
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

1.0 BACKGROUND

Licensees and applicants for operating licenses shall conduct a Detailed Control Room Design Review (DCRDR). The objective is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (NUREG-0660, Item I.D). Supplement 1 to NUREG-0737 requires each applicant or licensee to conduct a DCRDR on a schedule negotiated with the Nuclear Regulatory Commission (NRC).

NUREG-0700 describes four phases of the DCRDR and provides applicants and licensees with guidelines for its conduct.

The phases are:

1. Planning
2. Review
3. Assessment and Implementation
4. Reporting.

A Program Plan is to be submitted within two months of the start of the DCRDR. Consistent with the requirements of Supplement 1 to NUREG-0737, the Program Plan shall describe how the following elements of the DCRDR will be accomplished.

TECHNICAL EVALUATION REPORT
OF THE
DETAILED CONTROL ROOM DESIGN REVIEW
FOR
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

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ATTACHMENT to ENCLOSURE 1

TECHNICAL EVALUATION REPORT
OF THE
DETAILED CONTROL ROOM DESIGN REVIEW
FOR
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
HOPE CREEK GENERATING STATION

January 3, 1985

Kenneth O. Harmon
L. Rolf Peterson

Lawrence Livermore National Laboratory
for the
United States
Nuclear Regulatory Commission

1. Establishment of a qualified multidisciplinary review team

2. Function and task analyses to identify control room operator tasks and information and control requirements during emergency operations

3. A comparison of display and control requirements with a control room inventory

4. A control room survey to identify deviations from accepted human factors principles

5. Assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected

6. Selection of design improvements

7. Verification that selected design improvements will provide the necessary correction

8. Verification that improvements will not introduce new HEDs

9. Coordination of control room improvements with changes from other programs such as SPDS, operator training, Reg. Guide 1.97 instrumentation, and upgraded emergency operating procedures.

A Summary Report is to be submitted at the end of the DCRDR. As a minimum, it shall:

1. Outline proposed control room changes

2. Outline proposed schedules for implementation

3. Provide summary justification for HEDs with safety significance to be left uncorrected or partially corrected.

The NRC will evaluate the organization, process, and results of the DCRDR. Evaluation will include review of required documentation (Program Plan and Summary Report) and may also include reviews of additional documentation, briefings, discussions, and on-site audits. In-progress audits may be conducted after submission of the Program Plan, but prior to submission of the Summary Report. Preimplementation audits may be conducted after submission of the Summary Report. Evaluation will be in accordance with the requirements of Supplement 1 to NUREG-0737. Additional guidance for the evaluation is provided by NUREG-0700 and Appendix A to Standard Review Plan (SRP) Section 18.1 of NUREG-0800. Results of the NRC evaluation of a DCRDR will be documented in a Safety Evaluation Report (SER) or SER Supplement.

Significant HEDs should be corrected. Improvements which can be accomplished with an enhancement program should be done promptly.

2.0 DISCUSSION

The Hope Creek Generating Station (HCGS), operated by the Public Service Electric and Gas Company (PSE&G), is now under construction. As required by Supplement 1 to NUREG-0737, a completed Detailed Control Room Design Review (DCRDR) is required before an operating license can be issued. The HCGS DCRDR is in-progress.

PSE&G submitted a DCRDR Program Plan for HCGS to the NRC on October 14, 1983. The HCGS Program Plan was reviewed by the NRC staff to compare the applicant's response to the requirements of Supplement 1 to NUREG-0737 and the guidance of NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1. NRC staff comments on the HCGS Program Plan were issued on January 25, 1984. This review of the HCGS Program Plan included general comments on addressing control room modifications and additions made or planned as a result of post-TMI actions and possible integration of lessons learned from Salem ATWS events. Implications of the Salem ATWS events are discussed in NUREG-1000 and required actions are described in Section 1.2., Post Trip Review - Data and Information Capability, of the enclosure to Generic Letter 83-28.

In addition to the general comments above, the staff provided specific comments on the nine DCRDR elements identified in Supplement 1 to NUREG-0737.

PSE&G submitted a DCRDR Summary Report for HCGS to the NRC on August 14, 1984. A NRC human factors engineering preimplementation audit of the HCGS DCRDR was conducted at the plant site near Salem, New Jersey, on November 13 through November 15, 1984. The audit was carried out by a team of NRC personnel from the Human Factors Engineering Branch (HFEB) and consultants from the Lawrence Livermore National Laboratory, Livermore, California.

2.1 DCRDR Review Team

Supplement 1 to NUREG-0737 requires the establishment of a qualified multidisciplinary review team. Guidelines for review team selection are found in NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1. NUREG-0700 guidelines state that support of the applicant's management is needed to provide to the DCRDR team all of the information, equipment, and categories of manpower needed to conduct a control room design review.

As stated in the NRC staff review of the HCGS Program Plan, the DCRDR team structure and plan of operation is clearly defined. The Summary Report states that only one change was made in the HCGS review team since the Program Plan was submitted. That was a change of the Bechtel CRDR Coordinator. This change did not affect the performance of the HCGS DCRDR.

The DCRDR Program Plan stated that the PSE&G Project Manager would have overall responsibility for all aspects of the DCRDR and its coordination with other programs. The Project Manager's responsibility is confirmed in the DCRDR Summary Report. Figure 5 in the Summary Report shows the HCGS CRDR project management organization.

PSE&G states that they have dedicated the necessary resources to the DCRDR to ensure success of the project. These resources include the participation of knowledgeable technical and management personnel from PSE&G, Bechtel, and Essex Corporation. Essex Corporation is the PSE&G human factors consultant for the HCGS DCRDR. PSE&G stated that in conducting the HCGS DCRDR they would comply with NRC guidance and would develop an auditable documentation package of DCRDR analyses and findings. Based on observations during the NRC preimplementation audit, the audit team found that the intent of NUREG-0700 has been met.

NUREG-0700 recommends that during the DCRDR adequate office, storage, and meeting space should be provided for the review team and for any part-time

consultants and specialists. Equipment needs (e.g., sound-level meters, light meters, and photographic equipment) should be determined. All necessary equipment should be provided for the review team use.

We believe that HCGS management made the decision to meet the guidelines of NUREG-0700 to provide suitable equipment and workspace for the DCRDR process. The NRC audit team observed that adequate clerical, reproduction, and other peripheral support services have been made available to the DCRDR review team.

NUREG-0700 guidelines recommend that methods of data management should be established before the DCRDR is commenced.

Information and data management involves:

- Providing the review team members with reference material such as panel layout drawings, control room floor plans, and piping and mentation drawings,

- Developing standard forms to be used for recording the results of the control room review,

- Establishing a system for recording, storing, and retrieving data during the control room review.

It is apparent that HCGS used methods of data management that meet the guidelines of NUREG-0700. Both the Program Plan and the Summary Report include information that is well organized and documented. The preimplementation audit verified that the scope and depth of the HCGS data management and document control system meets the intent of NUREG-0700 guidelines.

Based upon our reviews of the HCGS Program Plan and Summary Report, we concluded that PSE&G management made a clear commitment to support the DCRDR process and that the review team members had suitable expertise for the job. The NRC preimplementation audit confirmed that the HCGS review team satisfies the requirement of Supplement 1 to NUREG-0737 to establish a multidisciplinary review team to conduct a DCRDR. The audit also confirmed that PSE&G has provided adequate management support for the DCRDR process.

2.2 Function and Task Analyses

Supplement 1 to NUREG-0737 requires the applicant to perform systems function and task analyses (SFTA) to identify control room operator tasks and to identify control room operator information and control needs during emergency operations. Supplement 1 to NUREG-0737 recommends the use of function and task analyses that have been used as the basis for developing emergency operating procedures technical guidelines and plant-specific emergency operating procedures to define these needs.

The steps for a top-down systems function and task analysis identified in the NUREG-0700 guidelines are:

1. Identification of Systems and Subsystems
2. Identification of Operating Events for Analysis
3. Function Identification
4. Operator Task Identification and Analysis

Operator information and control needs should be determined independently from the existing CR design. The analysis should include the appropriate functions of plant safety-related systems and the emergency operating procedures (EOPs) that must be used to ensure that the plant can be efficiently and reliably operated by available personnel during emergency conditions.

HCGS met with the NRC on February 2, 1984, to discuss the HCGS task analysis methodology, and submitted a written clarification on April 10, 1984. The methodology discussed in that clarification and in the Summary Report is different from that described in the HCGS CRDR Program Plan.

The following documents served as information inputs to the HCGS task analysis:

- BWROG Emergency Procedure Guidelines (EPGs).
- HCGS EOPs, which track closely with the EPGs and provide plant specificity. HCGS EOPs are in flow chart form.
- A draft document prepared by the BWROG to define operator information needs for a Graphic Display System as a generic basis for SPDS development.

During the NRC preimplementation audit of the HCGS control room, a review of the SFTA documentation was conducted by the NRC audit team. The audit team found that PSE&G is using a SFTA method that has identified operator information and control requirements independently from the existing control room design. These information and control requirements have thus far only been identified to the parameter level. The PSE&G proposed a satisfactory methodology for completing the FTA to identify the required plant-specific display and control characteristics to the level of parameter values, units, ranges, accuracies, and tolerances.

PSE&G has indicated that they will complete their DCRDR SFTA by accomplishing the following:

- Tasks will be updated by using plant-specific technical guidelines, technical specifications and plant systems information. This update will also include any guidelines and information on radioactivity control.

- Information and control characteristics associated with plant variables and plant parameters such as units, range, accuracy, and response time, will be identified appropriately for each task.

- The characteristics needed for given parameters will be summarized so that all display and control characteristics can be determined.

PSE&G also stated that they will compare the identified display and control needs with the available control room instrumentation, record HEDs, assess any HEDs that are found, and determine appropriate corrective actions. They plan to document all SFTA findings in the First Supplement to the HCGS Summary Report prior to licensing.

It appears that HCGS intends to meet the requirements of Supplement 1 to NUREG-0737 and conform with the staff conclusions from the May 4, 1984, meeting on the use of the BWROG EPGs. When performed and documented, the SFTA steps that PSE&G stated they will use to identify plant-specific display and control characteristics are expected to meet the DCRDR requirement.

2.3 Comparison of Control and Display Requirements with a Control Room Inventory

Supplement 1 to NUREG-0737 requires the applicant to make a control room inventory and to compare the operator display and control requirements determined from the task analyses with the control room inventory to determine missing controls and displays. Guidance in NUREG-0700 also calls for a review of the human factors suitability of instruments and controls used to satisfy operator information and control requirements.

A component report was completed for each component in the HCGS control room. The complete set of these forms constitutes a record of the control room inventory and is available for reference. The sources of information for this task were the photomosaics and engineering drawings of the panel layout.

The HCGS verification of information and control requirements identified through the task analysis will be done in the control room. The component reports will be used as detailed references.

We find this method acceptable. We conclude that HCGS should be able to meet the requirement of Supplement 1 to NUREG-0737 to compare the operator information and control needs with a control room inventory and identify missing controls and displays. However, until the SFTA is completed to identify the required plant-specific display and control characteristics, a comparison of information and control needs with the control room cannot be made. PSE&G must confirm that this comparison is made after completing the SFTA and identifying information and control requirements.

2.4 Control Room Survey

Supplement 1 to NUREG-0737 requires that a control room survey be conducted to identify deviations from accepted human factors principles. NUREG-0700 provides guidelines and criteria for conducting a control room survey.

The objective of the control room survey is to identify, for assessment and possible correction, characteristics of displays, controls, equipment, panel layout, annunciators and alarms, control room layout, and control room ambient conditions that do not conform to good human engineering practices.

A set of task plans, developed by Essex Corporation, was used to guide the survey data collection effort for the HCGS control room survey (CRS). These task plans cover completely all NUREG-0700 requirements. A criterion-by-criterion cross reference table between the task plan and NUREG-0700 is available. A task plan example has been provided as Appendix F in the Summary Report.

HCGS conducted their CRS between October 1983 and March 1984. The CRS was based on the application of the criteria from Chapter 6 of NUREG-0700. All

aspects of the CRS were completed at the time of the NRC audit except for the following survey items:

- HVAC
- Illumination
- Emergency Equipment
- Communications
- Computer Systems
- Ambient Noise

PSE&G by their letter of December 6, 1984, has committed to provide documentation of survey results to NRC of the HVAC, Illumination, Emergency Equipment, Communications, and Computer Systems surveys by November 1985. The results of an Ambient Noise survey will be documented and sent to the NRC no later than six months after fuel load.

The NUREG-0700 guidelines recommend that a review of operating experience be performed that includes the examination of available operating experience documents and a survey of control room operating personnel. HCGS intends to conduct an operating experience review that meets the guidelines of NUREG-0700. They have had operator interviews and have reviewed LERs from Peach Bottom 2 and 3. The LER review process began with a set of LERs that was "filtered" down to a set of LERs which were attributed to procedure design and another set attributed to operator error. The overall selection process resulted in a set of LERs which had possible application to the HCGS control room. A final analysis of these LERs resulted in no HEDs being identified as applicable to HCGS.

HCGS originally planned to conduct the operator survey in two parts: preliminary interviews, which were conducted during the first stage of the DCRDR; and a second set of interviews, which will be conducted after the plant is operational. In response to an NRC inquiry about the degree of operator input into the HCGS DCRDR, PSE&G decided to have a preliminary operator

interview included in the Summary Report. The purpose of this preliminary survey is to supplement, not replace, the more detailed operating personnel survey to be conducted after operating experience is acquired.

During the preimplementation audit, the NRC audit team reviewed the control room survey and operating experience review documentation. A comparison of selected CRS checklists, with actual control panels was made. The audit team reviewed the disposition of all noncorrective HEDs. The NRC audit team found that the HCGS CRS has been executed with diligence and that the CRS has defined substantive HEDs. The HCGS CRS acceptably meets the intent of NUREG-0700 guidelines and the requirements of NUREG-0737 Supplement 1 to conduct a control room survey.

2.5 Assessment of HEDs

Supplement 1 to NUREG-0737 requires that HEDs be assessed to determine which HEDs are significant and should be corrected. NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1, contain guidelines for the assessment process.

All HEDs that were identified during the HCGS review process were submitted to the HCGS Technical Advisory Team (TAT) to determine the implications of the HED with regard to safety and operational consequences.

The HCGS HED assessment process has two functions. First, it determines whether an HED should be corrected. Second, it determines when the correction should be implemented. Both decisions are determined by the category assignment of the HED.

There are four categories in which an HED can be placed. The determination of whether an HED should be corrected is a function of the consequences of an error due to the discrepancy and the estimated probability that such an error will occur.

Category I HEDs are those that describe an error that has actually been documented, such as in a LER. This category does not presently apply to HCGS since it is not operating and no pertinent LERs were found in the review of operating experience.

Category II HEDs are those judged to have potential safety consequences and some likelihood for error. This process was subjective since plant documentation determining acceptable levels for safety have not been completely developed. The TATs experience with similar specifications was used as guidance in evaluating the potential consequences of an HED.

Category III HEDs are those with potential safety consequences and a likelihood of error that was estimated to be very low. Also, HEDs not associated with safety consequences that could have an operational impact on plant availability and HEDs whose correction would provide significant benefit to the operator were placed in Category III.

All other HEDs were placed in Category IV.

In Appendix B of the Summary Report, there are a significant number of HEDs for which no corrective action is planned. These HEDs are also listed in Table 5.2 of the HCGS Summary Report. The NRC audit team reviewed these HEDs during the preimplementation audit generally agreed with the proposed resolutions. The audit team found that PSE&G has used an assessment method that assures an acceptable resolution to these HEDs with the exceptions stated below.

By their December 12, 1984, letter, PSE&G committed to resolution of HED A69 (zone markings) by doing a study of zone markings and reporting the results to the NRC one year after fuel load. PSE&G will revise the resolution of HED A140 (tagging) to indicate that use of plastic covers to block actuation of controls during maintenance are part of the tagging procedure.

During the preimplementation audit, a review of the applicant's HED assessment method, as described and reported, verified that it met the intent of the guidelines of NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1. An audit review of the HCGS documentation also determined that PSE&G has met the requirement of Supplement 1 to NUREG-0737 to assess HEDs to determine which HEDs at HCGS are significant and should be corrected.

2.6 Selection of Design Improvements

Supplement 1 to NUREG-0737 requires the selection of control room design improvements that will correct the significant HEDs. It also states that improvements that can be accomplished with an enhancement program should be done promptly.

The HCGS Summary Report states that all HEDs which were designated by the assessment process to receive a correction were first examined for the possibility of correction by surface enhancement. HEDs that could not be corrected by surface enhancements were analyzed for design alternatives.

All HEDs considered for design alternatives were reanalyzed. All information that contributed to the HED Report, such as operator interviews or the CR validation process, were reviewed to provide a clear definition of the functions, tasks, and operator requirements that might be affected by the discrepancy. This review process was used to determine and recommend a corrective action or alternate solutions for each HED. It was also intended to assure that no new HEDs are created.

The HCGS Summary Report states that in a few cases neither enhancement nor design change offered an appropriate solution. In those cases, a procedural or training solution was recommended.

A review of the documentation of the PSE&G selection of design improvements during the preimplementation audit confirmed that the applicant's methodology

is acceptable as described and reported. The NRC audit team determined that HCGS has met the requirement of Supplement 1 to NUREG-0737 to select design improvements that will correct the significant HEDs.

2.7 Verification of Control Room Design Improvements

Supplement 1 to NUREG-0737 requires verification that selected design improvements provide the necessary corrections and will not introduce new HEDs into the control room.

PSE&G described a method in their Summary Report, that is a part of the HCGS assessment process, to ensure that the design improvements provide the necessary corrections and will not introduce new HEDs into the control room. PSE&G planned to turn implementation of design improvements over to their normal engineering design process after selection of design improvements. They planned follow-up verification of installed improvements by their quality assurance staff.

PSE&G committed their December 6, 1984, letter to NRC that they will conduct a review of the as-built condition of control room design improvements.

When PSE&G completes their HED design improvement verification using acceptable techniques, including an auditable documentation trail of design changes and review by the DCRDR team, the design improvement verification requirement of Supplement 1 to NUREG-0737 will be satisfied.

2.8 Coordination of the DCRDR with Other Programs

Supplement 1 to NUREG-0737 requires that control room improvements be coordinated with changes from other initiatives such as SPDS, operator training, R.G. 1.97 instrumentation, and upgraded EOPs.

PSE&G described in Section 6.0 of the Summary Report their method to integrate (coordinate) the DCRDR with other programs. When fully implemented, they expect that this coordination effort will meet the coordination requirement of Supplement 1 to NUREG-0737.

The NRC audit team reviewed coordination of the DCRDR with other control room improvements during the preimplementation audit. The audit team observed that positive coordination was occurring.

As a result of the preimplementation audit, we believe that PSE&G will comply with the requirement of Supplement 1 to NUREG-0737 to coordinate the DCRDR with other control room improvement programs.

2.9 DCRDR Schedule

NUREG-0700 recommends that the planning of the control room review include the development of a detailed schedule of review tasks. NUREG-0700 also provides guidelines for determining the implementation schedule for design improvements.

Section 5.0 in the HCGS Summary Report describes an implementation schedule for design solutions. The discussion includes:

- A description of the improvements that have been initiated prior to the Summary Report.
- A list of HEDs that are planned to be corrected.
- An implementation schedule for these corrections.
- A list of HEDs for which no correction is planned.

PSE&G states that the HCGS engineering will be completed in December 1984 and implementation of the changes will be completed in May 1985, six months prior to fuel load. The time period for completion of CR changes was chosen to coordinate with the schedule for finalizing the EOPs.

HCGS included in their Summary Report an updated DCRDR schedule. It lists in Fig. 6 of the Summary Report the HCGS CRDR project milestones, and shows that the DCRDR will be completed with the submittal of a supplement to the Summary Report six months after fuel load.

During the preimplementation audit, the schedules for implementation and verification of control room design improvements were reviewed and discussed. Additional DCRDR schedule modifications and commitments were made in the PSE&G letter of December 6, 1984. We conclude that PSE&G has met the intent of NUREG-0700 in developing the HCGS DCRDR schedule and conducting their DCRDR according to that schedule.

3.0 CONCLUSIONS

Based upon our review of the HCGS Program Plan, DCRDR Summary Report, and the on-sit preimplementation audit, we believe that PSE&G is conducting a Hope Creek DCRDR that will meet the requirements of Supplement 1 to NUREG-0737 and the guidelines of NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1.

The HCGS DCRDR has yet to complete the requirements of Supplement 1 to NUREG-0737 for the following items:

1. The performance of system function and task analyses to identify plant-specific display and control characteristics to meet operator information and control needs.

2. The comparison of display and control requirements determined by function and task analyses with a control room inventory.

3. The verification that selected control room design improvements provide the necessary corrections and will not introduce new HEDs into the control room.

The audit team recommends that open control room survey items be completed and reported to the NRC by November, 1985, according to the commitments PSE&G made in their December 6, 1984, letter to the NRC. The detailed operating personnel survey, to be conducted after operating experience is acquired, and the final noise survey results should be reported no later than six months after fuel load, as stated in the PSE&G letter of December 6, 1984.

We believe that when these items are completed and documented, PSE&G will have conducted a HCGS DCRDR that meets all the requirements of Supplement 1 to NUREG-0737 and the guidelines of NUREG-0700 and NUREG-0800, Appendix A to SRP Section 18.1.

4.0 REFERENCES

1. Letter to A. Schwencer, NRC, from R. L. Mitti, submitting the Hope Creek Generating Station Detailed Control Room Design Review Program Plan, October 14, 1983.

2. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," May 1980; Revision 1, August 1980.

3. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, December 1982 (Generic Letter No. 82-33).

4. NUREG-0700, "Guidelines for Control Room Design Review," September 1981.

5. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 18.1, Appendix A, "Evaluation Criteria for Detailed Control Room Design Reviews," September 1984.

6. NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," April 1983.

7. Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.

8. Memorandum for Voss A. Moore, from S. H. Weiss, "Meeting Summary - Task Analysis Requirements of Supplement 1 to NUREG-0737 - May 4, 1984 Meeting with BWR Owners' Group Emergency Procedure Guidelines and Control Room Design Review Committees," May 1984.

9. Letter to A. Schwencer, NRC, from R. L. Mitti, submitting the Hope Creek Generating Station Detailed Control Room Design Review Summary Report, August 14, 1984.

10. Letter to A. Schwencer, NRC, from R. L. Mitti, submitting the Hope Creek Generating Station "Detailed Description of HCGS Function and Task Analyses Methodology," February 2, 1984.

APPENDIX J

ERRATA TO HOPE CREEK GENERATING STATION
SAFETY EVALUATION REPORT

Since the issuance of the Safety Evaluation Report, a number of SER sections have been identified in which either typographical errors exist or sufficient clarity is lacking. The following changes attempt to resolve these two occurrences. In no way do the following changes affect the staff's conclusions contained within the SER.

Page	Line No./Item	Change
xvii	37	Change "APPENDIG G" to "APPENDIX G".
1-8	TMI Item II.D.1	Change "position indication" to "performance testing".
1-11	Issue 9	Change "9.5.1.5" to "9.5.1.4".
2-17	6	Change "3.2" to "2.6".
	7	Change "34.3" to "35.4".
	8	Change "1" to "2".
	9	Change "31.0" to "30.0".
2-20	7-10	Delete last sentence.
3-27	15	Change "100" to "190".
5-3	37, 38	Delete "low pressure" before "core spray systems".
5-6	31	Change "emergency core cooling systems" to "core spray systems".
	33	Delete "core spray,".
5-11	30	Change "10-mCi/cc" to "10^{-6} mCi/cc".

Page	Line No./Item	Change
6-18	3	Change "272" to "278".
	10, 11	Change "two FRVS vent fans take 37 sec to reach full flow from time zero" to "a fourth FRVS recirculation fan takes 37 sec to reach full speed from time zero. Also, one FRVS vent fan takes 34 sec to reach full speed from time zero."
6-22	24	Change "Class 1" to "Class 1 and 2".
6-33	39	Change "on" to "and".
6-42	31	Change "and continues to maintain a reduced pressure (-0.25 psid)" to "to establish and maintain a reduced pressure (-0.25 WG)".
7-7	23	Change "B21-N0B0C" to "B21-N080C".
7-37	22, 23	Insert a period after CST and delete remainder of sentence.
7-38	8	Change "panels" to "panel".
7-40	15-20	Delete paragraph.
7-52	10, 13	Change "7.5.2.4" to "7.5.2.3".
9-9	33	Delete "auxiliary".
	34, 39	Change "enclosure" to "building crane".
	36	Change "fuel handling" to "refueling".
9-27	22	Delete "CAE" from list.
	28	Before last sentence of paragraph, add "The CAE is not safety related except for the redundant isolation dampers which fail closed."
	29	Add to end of last sentence "except for the CREF system. Failure of an operating CREF is alarmed in the control room. The redundant CREF is manually started."

Page	Line No./Item	Change
9-28	19 20	Change "two 50%" to "one 100%". Change "coils" to "coil".
9-32	9	Change "a low- and a high-efficiency filter" to "and a low-efficiency filter".
9-47	15	Delete "security system".
9-48	9	Delete "security".
9-52	14	Change "9.5.3.5 stated" to "9.5.3 did not indicate".
9-53	20	Change "Cree" to "Creek".
9-54	33	Change "or" to "and".
10-4	20	Change "Quality Group A criteria" to "Quality Group A criteria up to and including the outboard MSIV and to Quality Group B criteria up to and including the block valves".
11-1	34	Change "Table 11.1-1" to "Tables 11.1-1 and 11.1-3 through 11.1-6".
11-13	Control room ventilation	Change the range from "10^{-3} to 10^{-2}" to "10^{-7} to 10^{-4}".
12-8	19	Change "108" to "10^{-8}".
12-12	6, 7	Change "10-3" to "10^{-3}" and "104" to "10^{-4}".
15-5	30	Insert "and the feedwater turbine stop valves" in the parenthetical remark.
16-2	Issue 18	Delete Issue 18 because it is duplicated by Issue 16.
16-3	Issue 29 Issue 43	Change "7.7.2.3" to "7.7.2.2". Delete Issue 43.

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13. ABSTRACT (200 words or less)

Supplement No. 1 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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