NUREG-1048
Supplement No. 3

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

October 1985

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ABSTRACT

Supplement No. 3 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company on its own behalf as co-owner and as agent for the other co-owner, the Atlantic City Electric Company, 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 (PSE&G) (applicant) on its own behalf as co-owner and as agent for the other co-owner, the 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. Supplement Nos. 1 and 2 to the SER were issued in March 1985 and August 1985, respectively. The purpose of this supplement to the SER is to provide the staff evaluation of open items that have been resolved and to report on the status of all open items.

During its 296th meeting on December 13-15, 1984, the Advisory Committee on Reactor Safeguards 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 (reproduced as Appendix H in Supplement No. 1), 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.

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 and lists letters between the NRC staff and the applicant in chronological order. 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 N is a plant-unique analysis report by Brookhaven National Laboratory. Appendix O is the staff's analysis of the applicant's request to eliminate arbitrary intermediate pipe breaks in high energy piping.

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 available 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.

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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 Table 1.1 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 Table 1.2 and discussed further in the indicated sections of this report. If the staff review is completed for an item, it is identified as "closed."

1.9 License Condition Items

There are certain issues for which a license condition may be desirable to ensure that staff requirements are met by a specified date (Table 1.3). These conditions will be in the form of a condition in the body of the operating license.

Table 1.1 Outstanding issues (revised Table 1.2 from Supplement No. 2)

Issue	Status	SER section(s)
(1) Riverborne missiles	Under review	
(2) Equipment qualification	Partial closure	3.10 (also 3.10, Supplement 2)
(3) Preservice inspection program	Confirmatory	5.2.4, 6.6
(4) GDC 51 compliance	Closed	6.2.7 (Supplement 2)
(5) Solid-state logic modules	Under review	
(6) Postaccident monitoring instrumentation	Closed	7.5.2.3 (Supplement 2)
(7) Minimum separation between non-Class 1E conduit and Class 1E cable trays	Closed	8.3.3.3.3
(8) Control of heavy loads	Closed	9.1.5 (Supplement 1)
(9) Alternate and safe shutdown	Partial closure	9.5.1.4 (Supplement 2)
(10) Delivery of diesel generator fuel oil and lube oil	Closed	9.5.4.2 (Supplement 1)
(11) Filling of key management positions	Awaiting information	
(12) Training program items		
(a) Initial training programs	Closed	13.2.1.1 (Supplement 2)
(b) Requalification training programs	Closed	13.2.1.2 (Supplement 2)
(c) Replacement training programs	Closed	13.2.1.3 (Supplement 2)
(d) TMI issues I.A.2.1, I.A.3.1, and II.B.4	Closed	13.2.1.4 (Supplement 2)
(e) Nonlicensed training programs	Closed	13.2.2 (Supplement 2)

Table 1.1 (Continued)

Issue	Status	SER section(s)
(13) Emergency dose assessment computer model	Under review	
(14) Procedures generation package	Under review	
(15) Human factors engineering	Under review	

Table 1.2 Confirmatory issues (revised Table 1.3 from Supplement No. 2)

Issue	Status	SER section(s)
(1) Feedwater isolation check valve analysis	Closed	3.6.2
(2) Plant-unique analysis report	Partial closure	6.2.1.7
(3) Inservice testing of pumps and valves	Under review	
(4) Fuel assembly accelerations	Closed	4.2 (Supplement 2)
(5) Fuel assembly liftoff	Closed	4.2 (Supplement 2)
(6) Review of stress report	Closed	5.2.1.1
(7) Use of Code cases	Closed	5.2.1.2 (Supplement 2)
(8) Reactor vessel studs and fasteners	Closed	5.3.1.5
(9) Containment depressurization analysis	Under review	
(10) Reactor pressure vessel shield annulus analysis	Closed	6.2.1.5
(11) Drywell head region pressure response analysis	Closed	6.2.1.5
(12) Drywell-to-wetwell vacuum breaker loads	Closed	6.2.1.7
(13) Short-term feedwater system analysis	Closed	6.2.3
(14) Loss-of-coolant-accident analysis	Closed	6.3.5, 15.9.3 (Supplement 2)
(15) Balance-of-plant testability analysis	Under review	
(16) Instrumentation setpoints	Under review	
(17) Isolation devices	Awaiting information	
(18) Regulatory Guide 1.75	Under review	

Table 1.2 (Continued)

Issue	Status	SER section(s)
(19) Reactor mode switch	Closed	7.2.2.9
(20) Engineered safety features reset controls	Under review	
(21) High pressure coolant injection initiation	Closed	7.3.2.9
(22) IE Bulletin 79-27	Closed	7.4.2.1
(23) Bypassed and inoperable status indication	Under review	
(24) Logic for high pressure coolant injection interlock circuitry	Awaiting information	
(25) End-of-cycle recirculation pump trip	Closed	7.6.2.4
(26) Multiple control system failures	Closed	7.7.2.1
(27) Relief function of safety/relief valves	Closed	7.7.2.2
(28) Main steam tunnel flooding analysis	Closed	8.3.3.1.4
(29) Cable tray separation testing	Closed	8.3.3.3.2
(30) Use of inverter as isolation device	Closed	8.3.3.3.4
(31) Core damage estimate procedure	Closed	9.3.2
(32) Continuous airborne particulate monitors	Closed	12.3.4.2
(33) Qualifications of senior radiation protection engineer	Closed	12.5.1 (Supplement 2)
(34) Onsite instrument information	Closed	12.5.2
(35) Airborne iodine concentration instruments	Closed	12.5.2
(36) Emergency Plan items	Under review	
(37) TMI Item II.K.3.18	Closed	15.9.3 (Supplement 2)

Table 1.3 License conditions (revised Table 1.4 from SER)

License condition	Status	SER section
(1) Turbine system maintenance program		
(2) NUREG-0803 implementation		
(3) Inservice inspection		
(4) Postaccident sampling system	Removed	9.3.2
(5) Solid waste process control program		
(6) Partial feedwater heating		
(7) Cask drop accident	Removed	15.7.5

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, EQUIPMENT, AND COMPONENTS

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Location and Dynamic Effects Associated With the Postulated Rupture of Piping

In Section 3.6.2 of the SER, the staff identified a confirmatory issue regarding the dynamic analysis of the feedwater isolation check valves for the effects of a postulated pipe break in the feedwater piping outside containment. In letters dated July 26 and September 9, 1985, the applicant provided the results of the analyses of the feedwater check valves. The applicant's analyses included (1) a thermal hydraulic analysis to determine the peak pressures upstream and downstream of the valve disc as wel. as maximum disc angular speed, (2) a sensitivity analysis to determine the break location and feedwater check valve selection that yields the most conservative stress results, and (3) a stress analysis of the valve components of the disc, hinge arm, hinge pin, valve body, and seat ring. On the basis of the results of the analyses, the applicant concluded that the maximum stresses in the valve components are below the stress allowable values of Appendix F of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and the structural integrity of the feedwater check valve is maintained.

On the basis of the results of the applicant's analysis confirming the ability of the feedwater isolation check valves to perform their intended function following a feedwater line break outside containment, the staff concludes that the applicant has provided a reasonable basis to conclude that the safety concerns raised in the SER confirmatory issue have been acceptably resolved. Thus, the staff considers the confirmatory issue resolved.

3.6.2.1 Elimination of Arbitrary Intermediate Pipe Breaks

By letters dated June 11, July 3, and August 9, 1985, the applicant requested to use an alternative approach to the guidelines of Standard Review Plan (SRP) Section 3.6.2, Branch Technical Position MEB 3-1 (NUREG-0800), regarding the postulation of intermediate pipe breaks.

The staff has reviewed the applicant's request for the proposed deviation and believes that there is sufficient basis for concluding that an adequate level of safety exists to accept the proposed deviation. Appendix O to this supplement details the staff's review and conclusions concerning this topic.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

3.10.1.1 Introduction

Evaluation of the applicant's program for seismic and dynamic qualification of safety-related electrical and mechanical equipment consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general, and (2) an audit of selected equipment items to develop a basis for judging the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

Guidance for the evaluation is provided by SRP Section 3.10 and its ancillary documents, Regulatory Guides (RG) 1.100, 1.61, 1.89, and 1.92, NUREG-0484, and Institute of Electrical and Electronics Engineers (IEEE) Stds 344-1975 and 323-1974. These documents define acceptable methodologies for the seismic qualification of equipment. Conformance with these criteria is required to satisfy the applicable portions of General Design Criteria (GDC) 1, 2, 4, 14, and 30 of Appendix A to Title 10, Part 50, of the Code of Federal Regulations (10 CFR 50), as well as Appendix B to 10 CFR 50 and Appendix A to 10 CFR 100. Evaluation of the program is performed by a Seismic Qualification Review Team (SQRT), which consists of staff engineers and engineers from the Idaho National Engineering Laboratory (INEL, EG&G).

3.10.1.2 Discussion

The SQRT reviewed the equipment dynamic qualification information in Final Safety Analysis Report (FSAR) Sections 3.9.2 and 3.10 and made a plant site visit from May 7 through May 10, 1985. The purpose was to determine the extent to which the qualification of equipment, as installed at Hope Creek, meets the criteria described above. A representative sample of safety-related electrical and mechanical equipment, as well as instrumentation, included in both the nuclear steam supply system (NSSS) and balance of plant (BOP) scopes, was selected for the audit. Table 3.1 identifies the equipment audited. The plant site visit consisted of field observations of the actual, final equipment configuration and its installation. This was followed by a review of the corresponding design specifications, test, and/or analysis documents which the applicant maintains in the central files. Observing the field installation of the equipment is necessary to verify and validate equipment modeling employed in the qualification program. In addition to the document reviews and equipment inspections, the applicant presented details of the maintenance, startup testing, and inservice inspection programs.

3.10.1.3 Summary

On the basis of the observation of the field installation, review of the qualification documents, and responses provided by the applicant to SQRT's questions during the audit, the applicant's seismic and dynamic qualification program has been found to be well defined and adequately implemented. Upon closure of the issues identified in Table 3.1, and provided that the conditions delineated in the following sections are met, the seismic and dynamic qualification of safety-related equipment at Hope Creek will meet the applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100. These issues are to be resolved to the SQRT's satisfaction, and all the safety-related equipment must be completely qualified before fuel loading.

3.10.1.4 Generic Item

Review of the qualification documentation for the reactor core spray pump and motor (NSSS-1, MPL No. 1AP-206/E21-C001) revealed a questionable methodology used to demonstrate operability of the pump. The demonstration of adequate clearance during the safe shutdown earthquake was obtained by subtracting unsigned peak (square-root-of-the-sum-of-the-squares) displacements. The proper methodology is to subtract displacements at the modal level, then combine modal clearances according to any of the methodologies in RG 1.92. A spot check of two other pump analyses showed that questionable methodology was uniformly applied.

All equipment qualified by the response spectrum method, and which requires a clearance check, is affected. The applicant has agreed to review the methodology in question, revise calculations, and evaluate its impact where necessary.

3.10.1.5 Equipment-Specific Item

The required response spectra used in qualifying switches mounted in the standby liquid control (SLC) panel (NSSS-5, No. 145C3040P003) were obtained by calculations involving the floor response spectra and the dynamic characteristics of the panel. Because this methodology is rather new, a more detailed evaluation is required. From the review of a short summary and discussion during the audit, it appears that (1) the methodology is reasonable and (2) only two items in the plant (both on the SLC panel) were qualified using it. The applicant has provided such detailed evaluation in a June 11, 1985, submittal. It is being reviewed by the SQRT.

3.10.1.6 Confirmatory Issues

Confirmatory issues are:

(1) Inspection of the piping attached to the hydraulic control units (HCUs) (one of which was NSSS-2, MPL No. C11-0001) revealed several spans of piping that appeared to be inadequately supported. The applicant indicated that the lines had been shown to be adequate by dynamic analysis. However, an inhouse walkdown had previously initiated additional evaluation to confirm the acceptability of the subject piping. Results of the evaluation are to be transmitted to the NRC staff.

(2) During the inspection of the HCU piping discussed above, a gang support at vertical member F56722Q appeared to be complete, except that one line was not attached, leaving an unsupported span approximately 14 ft long. The applicant indicated that the design is adequate, but that a quality control verification has not been performed. The NRC staff will be informed of the results of the verification.

(3) The qualification report for the ITT actuator attached to the control damper (BOP-2, Tag No. 1HD-9603B1) consisted of a summary of testing performed at Wyle Laboratory by MCC Powers. For the qualification to be substantiated,

the original test report by the test laboratory is required. The applicant has provided the test report in the June 11, 1985, submittal. It is being reviewed by the SQRT.

(4) The ITT actuator for the control damper (BOP-2, Tag No. 1HD-9603B1) was found to be poorly supported in the field. The applicant indicated, on inquiry, that a support modification was required and produced an existing design change package (DCP-299) that specified the installation of an additional support for the actuator. Confirmation of the support installation will be provided to the NRC staff.

(5) A draft response to draft SER Open Item 103 (see letter from applicant dated August 20, 1984) was reviewed and found acceptable during the audit. The formal response has been included in Amendment 11 to the FSAR.

(6) Inspection of the reactor pressure vessel (RPV) level and pressure rack (NSSS-9, Tag No. 10C-026/H21-P026) revealed a poorly supported run of SST tubing attached to a panel-mounted device (1321-NOA50). The applicant indicated that the tubing had not been approved by either the engineering or the quality control department. Work was still in progress for this tubing; additional support will be provided in accordance with established acceptance criteria (Bechtel Power Corporation Specification 10855-J-S-1303). The NRC staff will be notified of the support installation.

3.10.2 Pump and Valve Operability Assurance

3.10.2.1 Introduction

The NRC staff performs a two-step review of each applicant's pump and valve operability assurance program to determine whether the program can ensure that all pumps and valves important to safety will operate when required for the life of the plant under normal and accident conditions. The first step is a review of Section 3.9.3.2 of the applicant's FSAR. However, this information is general in nature and lacks sufficient detail to determine the scope of the overall equipment qualification program as it pertains to pump and valve operability. The results of the FSAR evaluation appear as input to the SER. The resolution of any open SER issues is accomplished before or concurrently with the onsite audit.

A Pump and Valve Operability Review Team (PVORT), consisting of NRC staff engineers and engineers from INEL, EG&G, conducted the second step, which consisted of an audit of a representative sample of installed pump and valve assemblies and their supporting qualification documents at the plant site. On the basis of the results of both the audit and the FSAR review, the PVORT can determine whether the applicant's overall program conforms to the current licensing criteria in SRP Section 3.10. Conformance with SRP Section 3.10 criteria is required to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30, as well as Appendix B to 10 CFR 50.

The following sections include (1) a discussion of the PVORT review process, (2) the summary of PVORT findings concerning the applicant's overall pump and valve operability assurance program, (3) a discussion of the confirmatory issues resulting from the PVORT review, (4) Table 3.2, which summarizes the audit results, and (5) Table 3.3, which summarizes the pump and valve operability SER issues and their status.

3.10.2.2 Discussion

The PVORT reviewed the pump and valve operability assurance information in Section 3.9.3.2 of the Hope Creek FSAR and later conducted an onsite audit to determine the extent to which the pumps and valves important to safety meet the criteria listed above. The issues that resulted from the Hope Creek FSAR evaluation appeared in the SER. Several of these issues were adequately resolved by the applicant in an August 20, 1984, letter. The remaining SER issues were addressed and resolved during the onsite audit.

Table 3.3 summarizes the status of the six SER items originally identified. The staff believes that the applicant has adequately clarified his position concerning these items and agreed to appropriate commitments.

The onsite audit, which was conducted May 7 through May 10, 1985, consisted of field observations of the equipment configuration and installation for a representative sample of plant equipment. The PVORT selected four NSSS and six BOP pump and valve assemblies for evaluation. Table 3.2 summarizes the status of each assembly that was audited. The field observations were followed by a review of the design and purchase specifications, test/analysis documents, and other documents related to equipment operability, which the applicant maintains in his central files. In addition to reviewing information on the selected assemblies, the PVORT also reviewed other information on the plant's overall equipment qualification program. Included within this broad evaluation were those programs and procedures necessary to ensure that equipment qualification issues and concerns will continue to be addressed for the life of the plant. One such program, concerning the deep draft pump issue (refer to Office of Inspection and Enforcement (IE) Bulletin 79-15). was reviewed in depth.

The PVORT resolved all but two of the specific operability concerns that were identified during the audit. These two concerns involve (1) improper in-plant physical identification of the high pressure coolant injection (HPCI) turbine including manufacturing nameplate information and (2) service water pump's cyclone separators and instrumentation.

In addition, the applicant was informed of two other issues to which he must respond before fuel load. These two issues require the applicant to (1) evaluate any newly identified loads and (2) confirm that all pumps and valves important to safety are qualified. These concerns and issues are confirmatory in nature and form the basis for the discussion in Section 3.10.2.4.

The PVORT believes that the applicant is dealing with the equipment qualification issue in a positive manner. All of the SER items were adequately resolved on the basis of additional clarifications and appropriate commitments provided by the applicant. During the audit the applicant addressed all questions posed by the PVORT and committed to resolve certain unresolved issues before fuel load. Furthermore, the applicant discussed significant aspects of the overall equipment qualification program, such as preventive maintenance and vibration analysis. Consequently, the PVORT believes that the continuous implementation of the applicant's overall program should provide adequate assurance that the pumps and valves important to safety will operate as required for the life of the plant.

3.10.2.3 Summary

On the basis of the results of (1) the component walkdown and the review of the qualification document packages, (2) the additional explanations and information provided by the applicant throughout the audit, and (3) the resolution of the SER items, the staff concludes that an appropriate pump and valve operability assurance program has been defined and implemented. The continuous implementation of this overall program should provide adequate assurance that all pumps and valves important to safety will perform their safety-related functions as required for the life of the plant. On successful completion of the requirements delineated in the following section, the staff concludes that Hope Creek has qualified those pumps and valves important to safety so as to meet the applicable portions of GDC 1, 2, 4, 14, and 30 as well as Appendix B to 10 CFR 50.

3.10.2.4 Confirmatory Issues

On the basis of the PVORT's evaluation of the Hope Creek pump and valve operability assurance program, the staff has identified to the applicant the following confirmatory issues that must be completed before fuel load.

(1) At the time of the audit, the HPCI turbine identification including the manufacturer's nameplate information had been moved to another compartment with the turbine controller to meet environmental considerations. The applicant shall confirm that the turbine is properly identified and that the manufacturer's nameplate information has been affixed on the HPCI turbine itself.

(2) At the time of the audit, a problem concerning the cyclone separators on the station service water pump (AP-502) was identified. This problem involved failure of the cyclone separator system, which provides filtered cooling water to the service water pump bearings. The cyclone separator failure had gone undetected, apparently resulting in a service water pump packing failure. The applicant shall analyze this event to determine the precise cause of the failure and confirm that the service water pumps (a) can perform their safety function given the present cyclone separator system design and (b) are adequately instrumented to detect impending failures.

(3) The applicant shall confirm that the original loads used in tests/analyses to qualify pumps and valves important to safety are not exceeded by any new loads such as those imposed by a loss-of-coolant accident (hydrodynamic loads) or as-built conditions. If a new load exceeds that originally used, the impact of the new load on the qualification of the equipment must be assessed and reported to the NRC.

(4) At the time of the audit, approximately 10 to 15% of all pumps and valves important to safety had not been qualified. The applicant shall confirm that all pumps and valves important to safety are properly qualified and installed.

Table 3.1 Equipment audited

SQRT ID no.	Applicant ID no.	Equipment name and description	Safety function	Findings	Resolution	Status
BOP-1	1-EE-HV-4680	6-in. gate valve	Isolates suppression pool/torus water cleanup system from torus given a containment isolation signal			Qualified
BOP-2	1HD-9603B1	Control damper, duct mounted	Controls flow of outside air to diesel generator building	(1) Qualification report was a summary description. Original test report is required. (2) Damper's actuator was poorly supported because a support had not been installed (DCP-299).	(1) Test report will be submitted by June 11, 1985. (2) Confirmation of installation will be supplied by August 1, 1985.	Confirmatory
BOP-3	1EA-TPBC516-2	Termination panel	Supports transmission of electrical signals for station service water system			Qualified
BOP-4	1KJ-P17538D	Pressure gage	Provides local indication of pressure in standby diesel generator starting and control air tank			Qualified

Table 3.1 (Continued)

SQRT ID no.	Applicant ID no.	Equipment name and description	Safety function	Findings	Resolution	Status
BOP-5	1GQ-FSL9771D	Flow switch	Shuts down service water intake structure supply fan DV-503 when system low flow is sensed			Qualified
BOP-6	10B242	480-V motor control center	Provides power to Class 1E equipment			Qualified
BOP-7	1B-21F037E	Safety/relief valve (SRV) (vacuum breaker)	Relieves vacuum in an SRV line resulting from steam condensation			Qualified
BOP-8	1EC-TE46758	Resistance temperature detector	Is part of fuel pool pressure boundary			Qualified
BOP-9	10C-399	Remote shutdown panel	Allows safe shutdown of reactor remote from control room			Qualified
BOP-10(a)	HSS-4416B	Rotary switch on 10C-399	Transfers control of residual heat removal (RHR) pump BP202 from control room to remote shutdown panel 10C-399			Qualified
BOP-10(b)	TR-A201	Relay on 10C-399	Controls cooling suction inboard isolation valve HVF009			Qualified
BOP-10(c)	SRU-C399A1	Signal resistor unit on 10C-399	Converts 4-20 mA to 1-5 V dc signal for suppression pool level and reactor pressure RHR interlock			Qualified

Table 3.1 (Continued)

SQRT ID no.	Applicant ID no.	Equipment name and description	Safety function	Findings	Resolution	Status
BOP-10(d)	HS-F045	Switch on 10C-399	Controls reactor core isolation cooling turbine shutoff valve			Qualified
BOP-11	1ATB4207	Lube oil switch panel	Monitors diesel generator lube oil pressure; shuts down the generator and provides alarm signal on loss of pressure			Qualified
NSSS-1	1AP-206/E21-C001	Reactor core spray pump and motor	Supplies makeup water/ cooling to reactor during loss-of-coolant accident	A questionable methodology was used to ensure clearance in the pump during the safe shutdown earthquake.	Methodology will be reviewed, and all affected calculations will be revised and impact evaluated by August 15, 1985.	Open
NSSS-2	C11-D001	Hydraulic control unit	Operates scram valves on reception of a scram signal in order to force control rod insertion into the core	(1) Associated piping appeared flexibly supported. (2) A support for the associated piping (Vert. member F56722Q) appeared to be complete except that one line was not attached leaving approximately 14 ft of piping unsupported.	(1) Results of a previously initiated evaluation of adequacy of support for the piping will be submitted by August 15, 1985. (2) Results of a quality control verification of the calculation showing the subject piping to be adequate will be submitted by August 15, 1985.	Confirmatory

Table 3.1 (Continued)

SQRT ID no.	Applicant ID no.	Equipment name and description	Safety function	Findings	Resolution	Status
NSSS-3	1BFSVF18201/ C11F182	Control rod drive solenoid valve	Shuts off air to and vents air from scram discharge volume isolation valve air header			Qualified
NSSS-4	163C1303	Limit switch	Provides a signal to reactor protection system upon turbine valve closure			Qualified
NSSS-5	10C-011/H21-P011	Standby liquid control (SLC) panel	Monitors SLC system	In-cabinet spectra were generated using a new methodology.	Detailed description of the methodology will be submitted by June 11, 1985.	Open
NSSS-6	10P-216/E41-C002	High pressure coolant injection (HPCI) gland seal pump and motor	Is attached to a Class 1E bus; failure could short the bus			Qualified
NSSS-7	1BBV004/ B31F023B	Recirculation suction valve	Is part of reactor coolant boundary			Qualified
NSSS-8	163C1563	Pressure transmitter	Is a passive element in a Class 1E circuit and is part of a pressure boundary			Qualified

Table 3.1 (Continued)

SQRT ID no.	Applicant ID no.	Equipment name and description	Safety function	Findings	Resolution	Status
NSSS-9	10C-026/ H21-P026	Reactor pressure vessel level and pressure rack	Monitors reactor vessel level and pressure	A poorly supported run of tubing was found attached to a panel-mounted device. This deficiency was listed on the Construction Turnover Exception list.	Notification of installation of necessary supports will be made by August 1, 1985.	Confirmatory
NSSS-10	10C-6081/ H11-P608	Power range neutron monitor cabinet	Monitors reactor power level			Qualified
NSSS-11	10S-211/ E41-C002	HPCI turbine	Provides power to an HPCI pump			Qualified

Table 3.2 Summary of Pump and Valve Operability Review Team audit

Plant ID no.	Description	Safety function	Finding	Resolution	Status	Remarks
1AP-206 (NSSS)	Core spray pump	Supplies emergency cooling water to reactor core			Closed[1]	Specific concerns were resolved during audit.
1OP-217 (NSSS)	High pressure coolant injection (HPCI) booster pump	Enables HPCI pump to supply emergency cooling water to core	Note[2]	Note[3]	Closed[4]	The HPCI turbine nameplates were found attached to the control box. The control box had been relocated to a different room to meet environmental qualification requirements.
1AP-202 (NSSS)	Residual heat removal (RHR) pump	Supplies emergency cooling water to reactor core			Closed[1]	Specific concerns were resolved during audit.
1FV-4880 (NSSS)	HPCI turbine stop valve	Opens to supply steam to HPCI turbine			Closed[1]	Specific concerns were resolved during audit.
1AP-401 (BOP)	Diesel fuel oil transfer pump	Operates when required to fill diesel fuel oil day tank			Closed[1]	Specific concerns were resolved during audit.
1AP-502 (BOP)	Service water pump	Supplies cooling water to safety auxiliaries cooling system (SACS), turbine auxiliaries cooling system (TACS), and reactor auxiliaries cooling system (RACS) heat exchangers	Note[5]	Note[6]	Closed[7]	A failure of this component before the date of the audit led to this finding.

See footnotes at end of table.

Table 3.2 (Continued)

Plant ID no.	Description	Safety function	Finding	Resolution	Status	Remarks
1HV-4680 (BOP)	Torus water cleanup 6-in. gate valve	Closes to provide containment isolation			Closed[1]	Specific concerns were resolved during audit.
1HV-F015A (BOP)	RHR shutdown cooling 12-in. globe valve	Closes to provide containment isolation/open after accident to provide shutdown cooling			Closed[1]	Specific concerns were resolved during audit
1VH-2522B (BOP)	SACS-TACS 30-in. butterfly valve	Closes to isolate SACS loop B from loop A and TACS			Closed[1]	Specific concerns were resolved during audit.
1AE-V007 (BOP)	24-in. feedwater check valve	Closes to provide containment isolation			Closed[1]	Specific concerns were resolved during audit.
---	All pumps and valves important to safety	Operate as required during life of plant under normal and accident conditions	Note[8,9]	Note[10,11]	Closed[1]	None

[1] The qualification status is considered "closed," pending completion of Resolutions 10 and 11.
[2] No nameplate data were affixed to the HPCI turbine.
[3] The applicant shall confirm that nameplate data are properly affixed to the HPCI turbine.
[4] The qualification status is considered "closed," pending completion of Resolutions 3, 10, and 11.
[5] The applicant has not demonstrated that the design of the service water pumps' cyclone separators or instrumentation system is adequate.
[6] The applicant shall analyze this event to determine the precise cause of the failure and confirm that the service water pumps (a) can perform their safety function given the present cyclone separator system design and (b) are adequately instrumented to detect impending failures.
[7] The qualification status is considered "closed," pending completion of Resolutions 6, 10, and 11.
[8] The applicant has not verified that all new loads are enveloped by those loads originally used to qualify the equipment.
[9] The applicant has not completed the qualification of all pumps and valves important to safety.
[10] The applicant shall confirm that none of the new loads applicable to pumps and valves important to safety exceed those loads originally used to qualify the equipment.
[11] The applicant shall confirm that all pumps and valves important to safety are qualified before fuel load.

Table 3.3 Status of SER items for pump and valve operability assurance

SER items[1]	Finding/ resolution	Status
(1) There should be a list of equipment types that clearly shows the methods used for qualification. This list should also address which standards are met, in particular those cited in SRP Section 3.10.	Satisfactory	Closed[2]
(2) Clarification of how aging was incorporated in the qualification process should be contained in the FSAR. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant. The criteria for the maintenance and surveillance program should be contained in the FSAR.	Satisfactory	Closed[2]
(3) Identify in the FSAR which valves will be subjected to frequencies higher than 33 Hz (from hydrodynamic loads), and discuss the impact of these dynamic loads on valve qualification and performance.	Satisfactory	Closed[3]
(4) The FSAR should be amended to clearly show the loads and conditions considered in the qualification of safety-related pumps and valves.	Satisfactory	Closed[2]
(5) The extent to which draft standards ANSI/ASME QNPE-1 (N551.1), QNPE-2 (N551.2), QNPE-3 (N551.3), QNPE-4 (N555.4), and N41.6 and issued standard ANSI/ASME B.16.41 are used needs to be clearly stated in the FSAR. In addition, the applicant's position with respect to Regulatory Guide 1.148 must also be indicated in the FSAR.	Satisfactory	Closed[3]

See footnotes at end of table.

Table 3.3 (Continued)

SER items[1]	Finding/ resolution	Status
(6) The FSAR should be amended to show the extent to which operational testing is being used to meet the requirements of SRP Section 3.10. The extent to which operational testing is performed at full-flow and temperature conditions should be shown.	Satisfactory	Closed[3]

[1]The Hope Creek SER items for pump and valve operability assurance were identified in the SER.
[2]This item was adequately resolved on the basis of information submitted by the applicant in a letter dated August 20, 1984.
[3]This item was adequately resolved on the basis of information reviewed by the staff during the site audit on May 7-10, 1985. The applicant committed to close out this item in a manner and time frame that is acceptable to the staff.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With ASME Code and Code Cases

5.2.1.1 Compliance With 10 CFR 50.55a

As noted in Section 5.2.1.1 of the SER, the applicant in an attachment to the letter of January 30, 1979, identified those reactor coolant pressure boundary (RCPB) components of Hope Creek that are not in compliance with 10 CFR 50.55a. Additionally, the applicant identified the purchase order date and the component code applied in the construction of each component.

The staff in a letter dated November 4, 1981, requested the applicant to identify differences in the requirements between those editions of the ASME Code, Section III, used in the actual construction of RCPB components and the editions of the Code required for compliance with 10 CFR 50.55a on the basis of the construction permit date of November 4, 1974. This information was provided by the applicant in an attachment to a letter dated February 10, 1983.

The RCPB components that are not in compliance with 10 CFR 50.55a are the (1) reactor pressure vessel, including control rod drive housings, power range monitor in-core housings, and jet pump instrumentation penetration, (2) control rod drive, (3) main steam safety/relief valves, (4) main steam isolation valves, (5) main steam piping, (6) main steam flow elements, (7) reactor recirculation pumps, (8) reactor recirculation shutoff valves, (9) reactor recirculation bypass valves, and (10) reactor recirculation piping. The ASME Code, Section III editions and addenda used in the construction are those that were required at the time the components were procured and were based on a construction permit that was to be issued in 1971. Because the Hope Creek construction permit was not issued until November 1974, this delay has resulted in codes and standards identified that are different from those in 10 CFR 50.55a(c) through (f).

The staff's review of the differences between those codes and addenda used in the construction of RCPB components and the codes and addenda required by 10 CFR 50.55a indicated that there are two areas that require upgrading to the later code and addenda. These are (1) compliance with Article NA-3260, "Review of Stress Report," and (2) compliance with the fracture toughness requirements for older plants as defined in Branch Technical Position (BTP) MTEB 5-2, which is incorporated in SRP Section 5.3.2 (see Section 5.3.2 of the SER). It is the staff's position that other attempts to update the application to meet the requirements of 10 CFR 50.55a would not be compensated by an increase in the level of safety. The staff, therefore, approved the applicant's request for relief and acceptance of ASME Code, Section III, and addenda and the ASME Nuclear Pump and Valve Code and addenda applied in the construction of the RCPB components identified above. However, the staff required the applicant to comply with Article NA-3260 of the 1971 Edition of Section III of the ASME Code. Article NA-3260 requires that the owner or agent review each stress report provided by the component manufacturer to the extent necessary to determine that it has satisfied the requirements of the design specifications. The article further requires the owner or agent to certify that such a review has been conducted for each RCPB component and that the stress report does satisfy the requirements of the design specifications. In an attachment to the letter of July 19, 1985, the applicant has now verified that all RCPB components that are not in compliance with 10 CFR 50.55a have been reviewed in accordance with Article NA-3260 of the 1971 Edition of the ASME Code. The staff has reviewed the applicant's response and finds it acceptable.

The staff concludes that compliance with Article NA-3260 of the 1971 Edition of Section III of the ASME Code for each RCPB component will result in a component quality level that is commensurate with the importance of the safety function of the RCPB and constitutes an acceptable basis for satisfying the requirements of GDC 1, and is, therefore, acceptable. The staff considers SER Confirmatory Issue 6 resolved.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of U.S. Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory.

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g)

This evaluation supplements conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

The staff has reviewed the information in the FSAR through Amendment 11 dated July 1985; the Hope Creek Preservice Inspection (PSI) Program submitted on April 13, 1984; the results of the November 26, 1984, meeting with the applicant in Bethesda, Maryland (meeting summary dated December 18, 1985); the applicant's responses to the staff's request for additional information in letters dated February 14 and July 25, 1985; and Region I Inspection Report 50-354/85-28 dated July 19, 1985. The applicant states that, on the basis of the construction permit date of November 4, 1974, for Hope Creek, the PSI Program is required to meet ASME Code, Section XI, 1974 Edition with Addenda through Summer 1975. The applicant has voluntarily updated the PSI Program to ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978, except for the emergency core cooling (ECC) and residual heat removal (RHR) systems, which will be examined to the 1974 Edition, 1975 Addenda as required by the regulation. The use of later Code editions is acceptable as specified by 10 CFR 50.55a(g)(3). The staff has concluded that the examination sample in the PSI Program for systems and components within the reactor coolant pressure boundary is consistent with the applicable regulation and Code requirements.

The regulation requires that ASME Code, Class 2 piping in the ECC and RHR systems be examined. A volumetric examination of a representative sample of welds in these systems should be performed during the preservice examination. In a letter dated February 14, 1985, the applicant committed to perform an augmented volumetric examination to include 42 additional Class 2 welds in the RHR system and 36 additional welds in the reactor core spray system. The staff has reviewed this augmented examination and has determined that it is acceptable and satisfies the inspection requirements of GDC 36, 39, 42, and 45.

In the SER, the staff addressed the problem of ultrasonic examination of welds in the reactor recirculation system (RRS) with corrosion-resistant cladding (CRC) applied to the pipe inside diameter (ID) for corrosion reasons and to the outside diameter (OD) of the pipe for weld shrinkage or concentricity reasons. Hope Creek has approximately 57 welds with 308L stainless steel CRC that could affect the ultrasonic examination with conventional ultrasonic techniques. At the November 26, 1984, meeting, the applicant presented a progress report and demonstrated a 1.5-MHz dual-element angle beam transducer designed to provide a 45° refracted longitudinal wave with a convergent crossover depth at the approximate thickness of the 12-in. mockup of a recirculation piping riser. As a result of this meeting, the staff requested detailed information on the specific welds with CRC and typical dimensions of the cladding; a plant demonstration of the ability to detect ID cracks through OD cladding for all applicable pipe sizes was also requested.

The applicant has been engaged in a comprehensive program to develop appropriate instrumentation to examine piping welds with CRC and to obtain representative calibration standards and laboratory-cracked mockups. This effort was coordinated with other boiling-water reactor (BWR) owners with similar pipe configurations with CRC. In a February 14, 1985, submittal, the applicant provided detailed information on the specific welds with CRC and a drawing showing typical dimensions of the cladding for all applicable pipe sizes. During the week of June 10, 1985, NRC staff personnel witnessed a plant demonstration on the 12-in.-diameter pipe with CRC. On the basis of the demonstrated ability to detect cracks and the observed similarities of the acoustic noise levels in the samples and the production welds, the regional inspector concluded that the procedure was acceptable for the examination of 12-in.-diameter CRC piping welds at Hope Creek. For further information, see Inspection Report 50-354/85-28. Additional test specimens containing both cracks and notches were fabricated from 22-in. and 28-in. recirculation piping from Hope Creek Unit 2. Each specimen contains a CRC weld with cladding on both the inside and outside diameters. Another plant demonstration using these samples resulted in conclusions similar to those in Region I Inspection Report 50-354/85-28. In a submittal dated July 25, 1985, the applicant provided a summary report of the program to develop an effective procedure to examine CRC welds. The staff has reviewed all the information described above and concludes that the preservice ultrasonic examination of the CRC welds in the Hope Creek recirculation piping system is acceptable and provides a baseline for future examinations.

The applicant has docketed information to resolve the issues of the preservice volumetric examination of ASME Code, Class 2 pipe welds in the ECC and RHR systems and the preservice ultrasonic examination of pipe welds with CRC. Therefore, the staff considers the review of the Preservice Inspection Program a confirmatory issue contingent on the applicant's submittal of all requests for relief from impractical examination requirements with supporting technical justifications. The staff will report this evaluation in a future supplement to the SER.

The initial Inservice Inspection (ISI) Program has not been submitted by the applicant. The program will be evaluated after the applicable ASME Code edition and addenda can be determined on the basis of 10 CFR 50.55a(b) but before ISI commences during the first refueling outage.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials (Fracture Toughness)

5.3.1.5 Reactor Vessel Materials (Materials and Fabrication)

As noted in the SER, the staff's review of the FSAR concluded that the reactor vessel studs and fasteners satisfy most of the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." However, the FSAR, at the time the SER was issued, did not discuss the nondestructive examinations of the closure bolts and nuts. The staff needed confirmation that the Code-specified inspections were performed on those items. This was identified as an SER confirmatory issue.

By letter dated May 24, 1985, the applicant provided the needed confirmation. Additionally, the applicant amended the FSAR (Amendment 11) to include this information concerning the inspections and examinations performed. On the basis of the review of this material, the staff finds the applicant is in conformance with RG 1.65.

Integrity of the reactor vessel studs and fasteners is ensured by conformance with the recommendations of RG 1.65. Compliance with these recommendations satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR 50, as detailed in the provisions of ASME Code, Sections II and III. The staff considers SER Confirmatory Issue 8 resolved.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.5 Subcompartment Pressure Analysis

In Section 6.2.1.5 of the SER, the staff stated that it will verify that the calculated differential pressure for the various subcompartments (reactor pressure vessel shield annulus (SER Section 6.2.1.5.1) and the drywell head region (SER Section 6.2.1.5.2)) will not exceed the design values.

The applicant's subcompartment nodal models consider all major flow restrictions. The staff has reviewed the applicant's models and the results of the analyses of the differential pressure. On the basis of a comparison of the results provided by using similar analytical models for similar subcompartment configurations, the staff finds the applicant's analyses of the differential pressures resulting from the design-basis accidents to be conservative, and, therefore, acceptable.

In addition to the subcompartment differential pressure analysis, the applicant has performed force calculations on the reactor pressure vessel from the asymmetric loads calculated in the subcompartment analysis. The staff has reviewed the applicant's method of determining forces from the differential pressure results and finds these methods and results acceptable.

6.2.1.7 LOCA Pool Dynamics

In July 1980, the staff issued a report, NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program," to address the NRC acceptance criteria for the Mark I Containment Long-Term Program, which are intended to establish design-basis loads that are appropriate for the anticipated life of each Mark I BWR facility and to restore the originally intended design safety margins for each Mark I containment system.

Since the issuance of NUREG-0661, the Mark I owners submitted additional reports in which they provided additional justifications for the adequacy of (1) the data base for specifying torus wall pressure during condensation oscillations, (2) the consideration given to asymmetric torus loading during condensation oscillations, and (3) the effect of fluid compressibility in the vent system of pool-swell loads. As a result of the staff's and its consultant's (Brookhaven National Laboratory) evaluation of these reports, the staff issued Supplement No. 1 to NUREG-0661 on August 1982.

The applicant submitted a plant-unique analysis report (PUAR) on the pool dynamic loads for the Hope Creek Mark I containment. This report describes the specific application of the generic Mark I pool dynamic loads and methods and the plant-unique loads used in assessing the capability of the containment and components to accommodate the pool dynamic loading phenomena.

Brookhaven National Laboratory (BNL) was contracted to review the PUAR for compliance with the staff's acceptance criteria and to evaluate the acceptability of any proposed alternative load specification.

A summary of the BNL review and status for each of the pool dynamic loads is presented in Appendix N to this report. As indicated in Appendix N, the applicant has adopted all but a few of the generic criteria. For those few exceptions alternative criteria were proposed. The BNL evaluation of these criteria is included in the appendix. On the basis of its review, the staff endorses the BNL evaluation and conclusion.

In conclusion, the staff has completed an assessment of Hope Creek against its generic acceptance criteria. It has also reviewed those few areas where alternative criteria have been proposed. In addition, the staff has completed its review of those areas where additional information was relegated to the plant-unique review. In each of these areas, the staff has concluded that the pool dynamic loads utilized by the applicant are conservative and, therefore, acceptable.

In addition, the applicant submitted a response to Generic Letter 83-08, dated February 2, 1983, which was sent to all applicants and licensees for plants with Mark I containments. In the generic letter, the staff requested information on a potential failure mode of the drywell-to-torus vacuum breakers during the chugging and condensation oscillation phase of a loss-of-coolant accident (LOCA).

In the submittal the applicant indicated that he has followed the methodology and assumptions in the Continuum Dynamic, Inc. (CDI) Generic Report (CDI 84-3), which describes the models used to compute the vacuum breaker response to chugging and condensation oscillation events. The staff had previously reviewed and found acceptable the CDI report and transmitted its evaluation report to General Electric by letter dated December 24, 1984, from D. Vassalo. On the basis of its review of the applicant's submittals, the staff concludes that the analysis to predict the differential pressure load across the vacuum breaker and the corresponding actuation velocities were done using the generically acceptable methodology.

Therefore, the applicant's conclusions that the valve opening and closing velocities are within the design capabilities of the vacuum breaker valves is acceptable.

6.2.3 Secondary Containment Bypass Leakage Paths

In Section 6.2.3 of the SER, the staff noted that the applicant had committed to provide a cross-tie on the feedwater fill network. This cross-tie will permit the fill network to perform its intended safety function following a single active failure and will ensure the sealing function of this system for at least 30 days following a LOCA.

The applicant also committed to perform a confirmatory analysis to verify that the pressure in the feedwater system piping would be sufficient to prevent the outward leakage of radioactive containments through the isolation valves during the approximately 1-hour period after the accident before the water seal is reestablished between the isolation valves via the fill system.

The applicant has installed the cross-tie on the feedwater network and has performed an analysis of the feedwater system. This analysis indicates that a water seal will be maintained upstream of the third feedwater heater, which will prevent bypass leakage through the feedwater system until the feedwater fill system establishes a long-term water seal between the containment isolation valves.

The applicant has indicated that the feedwater piping from the reactor pressure vessel back to the in-line anchor outside containment is designed to seismic Category I standards. The piping from the in-line anchor to the condenser is not constructed to seismic Category I standards. However, this piping has been seismically analyzed to the "Uniform Building Code" (International Conference of Building Officials), which is above and beyond the requirements for typical nonseismic piping.

The staff, therefore, concludes that bypass leakage through the feedwater line will be eliminated for the following reasons:

(1) The applicant has provided a piping cross-tie on the feedwater line fill network, which will permit the fill network to perform its intended safety function assuming a single active failure and will ensure the sealing function for at least 30 days following a LOCA.

(2) The staff has reviewed and found acceptable the applicant's analyses that show that the feedwater piping pressure is sufficient to prevent outward leakage during periods when the fill system is not operable.

Therefore, the staff concludes no bypass leakage of the feedwater system is expected to occur.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

6.6.3 Evaluation of Compliance With 10 CFR 50.55a(g)

This evaluation supplements conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

The staff has reviewed the information in the FSAR through Amendment 11 dated July 1985; the Hope Creek Preservice Inspection (PSI) Program submitted on April 13, 1984; the results of the November 26, 1984, meeting with the applicant in Bethesda, Maryland; the applicant's responses to the staff's request for additional information in letters dated February 14 and July 25, 1985; and Region I Inspection Report 50-354/85-28 dated July 19, 1985. The applicant states that, on the basis of the construction permit date of November 4, 1974, for Hope Creek Unit 1, the PSI Program is required to meet ASME Code, Section XI, 1974 Edition with Addenda through Summer 1975. The applicant has voluntarily updated the PSI Program to ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978, except for the ECC and RHR systems, which will be examined to the 1974 Edition, 1975 Addenda as required by the regulation. The use of later Code editions is acceptable as specified by 10 CFR 50.55a(g)(3). The staff has concluded that the examination sample in the PSI Program for systems and components within the reactor coolant pressure boundary is consistent with the applicable regulation and Code requirements.

In a request for information, the staff indicated that the PSI Program did not identify the examination for the scram discharge volume (SDV) system. In the February 14, 1985, submittal, the applicant states that the SDV system is designed and fabricated to Nuclear Class 2 requirements and will be examined on the basis of the requirements of the 1977 Edition of Section XI as an ASME Code, Class 2 system. Because the revision to the PSI Program submitted on April 13, 1984, includes this information, the staff considers this issue resolved. Section 5.2.4.3 of this supplement describes the resolution of the volumetric preservice examination of a representative sample of welds in the ECC and RHR systems.

The staff, therefore, considers the review of the PSI Program a confirmatory issue contingent on the applicant's submittal of all requests for relief from impractical examination requirements with supporting technical justifications. The staff will report this evaluation in a future supplement to the SER.

The initial Inservice Inspection (ISI) Program has not been submitted by the applicant. This program will be evaluated after the applicable ASME Code edition and addenda can be determined on the basis of 10 CFR 50.55a(b), but before ISI commences during the first refueling outage.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Protection (Trip) System

7.2.2 Specific Findings

7.2.2.9 Reactor Mode Switch

Mode switch contact and mode switch operating mechanism malfunctions have caused inadvertent protective actions. Similar malfunctions could have rendered redundant channels of protective functions inoperable. IE Information Notice 83-42 provided notification of potentially significant events concerning mode switch malfunctions.

The reactor mode switch that was installed at Hope Creek was of the type that is susceptible to misoperation as described in IE Information Notice 83-42. This switch was replaced with a modified switch having an identical contact configuration and wiring scheme. The applicant provided the design details for the new mode switch, including an assessment of remote switch misoperations.

As documented in the FSAR, the conclusions from these assessments are that all misoperations of the mode switch are detectable by various means. Furthermore, the applicant has proposed a surveillance action associated with the average power range monitor and intermediate range monitor channel functional testing to ensure detection of mode switch misoperation.

On the basis of its review of the information provided, the staff finds that the applicant has adequately described the current mode switch design and performed an adequate failure analysis. Therefore, the staff concludes that the mode switch design is acceptable and Confirmatory Issue 5 as listed in SER Section 7.1.4.2 is resolved.

7.3 Engineered Safety Features Systems

7.3.2 Specific Findings

7.3.2.9 High Pressure Coolant Injection Initiation

High pressure coolant injection (HPCI) is initiated automatically on low water level in the reactor vessel (via relays K1A, K3A, K72, and K73) or on high pressure in the drywell (via relays K5A, K6A, K74, and K75). Contacts from these relays operate HPCI initiation relays K32, K33, and K34, which do not seal in. However, contacts from relay K32 operate the HPCI initiation seal-in relay (K35). Contacts from the seal-in circuit operate the initiation sealed-in system status indicator and all the required components for HPCI startup, except for the HPCI pump discharge valve that is operated by contacts from relays K32, K39, and K44. Relay K32 is upstream of the initiation seal-in circuit. The time required for the steam inlet valve to move sufficiently to satisfy the interlock (relay K39) was estimated to be 13 sec. Therefore, if the automatic initiation signal did not persist for 10-15 sec, the HPCI pump discharge valve would not open automatically.

The system-level initiation switch is wired in parallel with the low water level automatic contacts. The switch itself has no mechanical seal-in feature. Therefore, when the switch is released (before the 10 to 15 sec elapse), the initiation signal no longer exists, and the HPCI pump discharge valve does not open. The HPCI would come up to an active standby mode, with the flow being recirculated through the miniflow line. The staff concluded that this portion of the HPCI design did not conform to the IEEE Std 279 "go-to-completion" criterion.

IEEE Std 279 requires that the protective system automatically initiate the protective action of the appropriate safety system when a plant condition reaches a predetermined level. Manual initiation capability is also required. Once initiated, either automatically or manually, the initiation must cause an action, or a sequence of actions, that results in providing the necessary safety function. The go-to-completion criterion is satisfied typically with initiation seal-in circuitry.

The applicant had proposed replacing the K32 contact in the pump discharge valve (E41-F006) logic circuit with contacts from a relay that is sealed-in on HPCI initiation (whether automatic or manual). This was acceptable to the staff, and the staff concluded that this modified design would meet the requirements of IEEE Std 279. However, as a confirmatory issue, the staff required the applicant to submit the final drawings depicting the HPCI design.

By letter dated July 1, 1985, the applicant submitted the final drawings that depicted the modified HPCI initiation logic. On the basis of the results of its review of these drawings, the staff finds that the final drawings are correct and that the design of the initiation logic does seal-in in accordance with the criterion of IEEE Std 279. Therefore, Confirmatory Issue 7 as listed in Section 7.1.4.2 of the Hope Creek SER is resolved.

7.4 Systems Required for Safe Shutdown

7.4.2 Specific Findings

7.4.2.1 IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation"

As a result of an event involving the loss of a significant amount of control room information at the Oconee plant, the staff issued Office of Inspection and Enforcement (IE) Bulletin 79-27.

In response to this bulletin, the applicant had stated that an analysis will be conducted based on the Limerick Generating Station approach for answering the concerns raised in IE Bulletin 79-27. As discussed in SER Section 7.4.2.1, the staff had reviewed the applicability of the Limerick Generating Station approach to the resolution of the concerns raised in IE Bulletin 79-27 to Hope Creek and found it acceptable, subject to the review and approval of the recommended hardware or procedural changes or the justification for not requiring them.

As a confirmatory issue, the staff required the applicant to document the results of the analysis and to recommend hardware or procedural changes as appropriate in response to IE Bulletin 79-27. However, in a report by the applicant dated August 1984 and in a letter submitted to NRC on February 7, 1985, the applicant provided a somewhat revised methodology for performing the analysis and the results of the analysis performed.

An outline of this revised methodology follows:

(1) Identify the systems (and subsystems) required to bring the plant to a cold shutdown under emergency conditions.

(2) Identify the devices that provide information to the operator to achieve a cold shutdown.

(3) Identify the power supply buses associated with the devices in Item (2) above.

(4) Analyze the effect of a loss of power to each bus identified in Item (3) above, and determine the ability to achieve a safe shutdown with this bus loss.

(5) Review system drawings to determine what type of information is available to the operator to alert him/her to a bus loss.

(6) Review the Hope Creek emergency operating procedures, and verify that the procedures to restore power to the affected power buses are adequate.

(7) Review the final plant operating procedures, and make modifications if necessary.

On the basis of its review of the applicant's response to IE Bulletin 79-27, the staff concludes that there is reasonable assurance that any single instrumentation and control bus failure will not result in a plant condition requiring reactor shutdown, and simultaneously cause the failure of instrumentation relied on to achieve reactor shutdown.

In addition, the failure of each of the buses is annunciated and displayed in the control room. Furthermore, the analysis showed that the operator has alternative instruments and shutdown paths to achieve a cold shutdown condition.

However, the applicant had not provided a response to Item (5) of the staff's question regarding IE Bulletin 79-27. Item 5 discusses a re-review of IE Circular 79-02, which is required by Action 3 of IE Bulletin 79-27. When the staff expressed this concern to the applicant, the applicant stated that the solid-state inverters to be used at Hope Creek had been included as part of the study and that no problems existed with the Hope Creek inverter design. This information was submitted in a letter dated August 26, 1985. The staff finds the applicant's responses acceptable, and, therefore, Confirmatory Issue 8 as listed in SER Section 7.1.4.2 is resolved.

7.6 Interlock Systems Important to Safety

7.6.2 Specific Findings

7.6.2.4 End-of-Cycle Recirculation Pump Trip

Two redundant Class 1E actuation logics (trip system A and trip system B) are provided to initiate an end-of-cycle (EOC) recirculation pump trip (RPT) on either turbine stop valve closure or turbine control valve fast closure. Relay contacts from valve position instrument channels (one channel per valve) are arranged in a two-out-of-two energize to actuate trip logic in each division, for both the turbine stop and control valves.

The first-stage pressure transmitters for RPT A are on a different instrument sensing line from those associated with RPT B and are attached to the high-pressure turbine casing at different locations. By using this configuration, a sensing-line failure would only bypass one RPT breaker (A or B). In addition, a single failure of any transmitter will not preclude RPT operation because either logic division will trip both recirculation pumps (circuit breakers 3A and 3B for RPT A and 4A and 4B for RPT B).

The staff review of the elementary diagrams did not indicate that the EOC RPT transferred the pumps to low-frequency motor generator (MG) sets after tripping their main power supplies. On previously reviewed BWRs, this transfer takes place after the RPT and the pumps have run at approximately one-quarter their normal speed. The staff determined that there was not sufficient information to complete its review regarding the EOC RPT. As a confirmatory issue, the applicant was required to submit design details showing the transfer of the recirculation pump power supply to a lower frequency MG set upon EOC RPT.

By letter dated March 1, 1985, the applicant stated that the EOC RPT provides for the insertion of negative core reactivity to improve thermal margins for certain pressurization transients. The early part of the transient and the core void reactivity that the EOC RPT produces are not dependent on whether the final recirculation flow is determined by natural circulation or by a small power input to the recirculation pumps from a low-frequency MG set. The transfer to the low-frequency MG set is an inherent design characteristic of the BWR 5/6 plants but currently does not exist in BWR 4 plants.

The staff has verified the above information and has concluded that the EOC RPT transfer to the low-frequency MG sets will serve no safety function in a BWR 4 plant and its absence is not detrimental to the effectiveness of the EOC RPT design at Hope Creek. This resolves Confirmatory Issue 11 as listed in Section 7.1.4.2 of the SER.

7.7 Control Systems

7.7.2 Specific Findings

7.7.2.1 Multiple Control System Failures due to High Energy Line Breaks and Failures of Shared Components

A concern was raised in IE Information Notice 79-22 that if control systems are exposed to the adverse environment caused by a high energy line break (HELB), the systems may malfunction in a manner that would cause consequences more severe than those assumed in the safety analysis of Chapter 15 of the FSAR. In response to this concern, the applicant stated that an analysis will be conducted based on the General Electric methodology for answering the concerns raised in IE Information Notice 79-22. The methodology ensures a systematic, comprehensive analysis of HELBs and the consequential control system failures. An outline of this methodology follows:

(1) Identify all non-safety-grade control systems and components within these systems whose failure could affect the critical reactor parameters of water level, pressure, and power.

(2) Establish assumptions and criteria for determining high energy lines and pipe break locations and for evaluating the consequences (pipe whip, jet

impingement, environment) of pipe breaks. Environmental conditions such as high temperature, high pressure, and high humidity will be considered.

(3) Identify from appropriate plant drawings those plant locations where high energy lines with postulated break locations coexist with non-safety components of control-grade systems.

(4) Conduct a plant walkdown to verify the locations of control system components and to determine their proximity to HELB locations.

(5) Postulate pipe breaks in the zones defined, and determine which control system components are affected by each possible pipe break.

(6) Analyze the potential effects on the control system components impacted, and determine the effects on any controlled component.

(7) Combine the effects of the HELB with potential simultaneous malfunctions of adjacent control system components, and determine the effect on the critical reactor parameters.

(8) Compare the effects with the transient and accident analyses in Chapter 15 of the FSAR, considering an additional single active component failure in a mitigating safety system.

(9) Identify postulated events that are beyond Chapter 15 analyses, and recommend corrective actions.

The applicant provided the results of this analysis in a letter dated August 24, 1984. The results indicate that the applicant has analyzed the worst-case combined efforts of each HELB and all consequential non-safety-related/control system failures. In each case all failure modes and their consequences were analyzed. The consequences of these events were then compared with the accident and transient analyses in Chapter 15 of the Hope Creek FSAR. The analyses for several HELB zones uncovered an accident scenario that is not specifically addressed in the Chapter 15 analyses. In particular, by reducing the feedwater temperature, a delayed turbine trip could be initiated at a power level higher than that assumed in FSAR Section 15.2.3. The worst-case transient is a turbine trip at a power level just below the high thermal power monitor scram setpoint. A computer analysis was performed for this worst-case transient, which used the initial conditions, assumptions, and computer codes identified in FSAR Chapter 15. However, subsequent analysis performed by the applicant has demonstrated that the effects of this accident event, including consideration of a single active failure in a mitigating safety system, are bounded by the Chapter 15 analyses. The applicant has determined that the combined consequences of all other HELBs and consequential non-safety-related/control system component failures are also bounded by the Hope Creek accident and transient analyses in Chapter 15 of the FSAR.

On the basis of a detailed review of the applicant's analysis of HELBs and consequential non-safety-related/control system component failures for several different zones (including the worst-case-event zone), the staff has concluded that the methodology used and the results of the analysis performed by the applicant are acceptable, and, therefore, this concern is resolved.

A concern has also been raised that if several control systems or control and safety systems are supplied information from common sensors (including headers or impulse lines) or are supplied power from a common power source, a failure of the power source, or sensors, or a rupture/plugging of a header or impulse line could cause multiple control system failures not bounded by the safety analysis in Chapter 15 of the FSAR. In response to a verbal question from the staff, the applicant stated that an analysis will be conducted based on the General Electric methodology to answer NRC concerns about common power source failures and common sensor failures. The methodology is systematic and comprehensive and examines control system interactions to establish the limiting-case events.

The outline of the methodology for the common power source analysis follows:

(1) Identify all non-safety-grade control systems that have the potential for affecting the critical reactor parameters of water level, pressure, or power.

(2) Review these control systems at the component level, and identify the effects of the loss of power on each system component and the subsequent interactions with other components and systems.

(3) Generate bus trees denoting the bus hierarchy and cascading configuration of all power buses that supply components of the control systems under study.

(4) Perform a combined effects analysis. Evaluate the failure of each power bus (e.g., load center, motor control center) starting with the lowest level source common to multiple control systems and working up each bus tree to the highest common power level. At each level examine the effects of the single bus failure and the consequences of cascading bus failures on all control system components.

(5) Postulate the limiting transient events as a result of the combined effects analysis, and compare these events with those analyzed in Chapter 15.

(6) Perform additional transient calculations or analyses necessary to ensure that the worst-case limiting event is bounded by those analyzed in Chapter 15 with the assumption that there is a single active failure in a safety system required to mitigate the effects of the event.

(7) Document the results of the analyses of common power source failure, and provide recommendations as appropriate.

The outline of the methodology for the common sensor and sensor line failure analysis follows:

(1) Identify the non-safety-grade control systems to be included as in Item (1) of the methodology for the analysis of common power source failures.

(2) Identify all instrument sensing lines and sensors utilized by two or more of these control systems.

(3) Analyze the effects of a complete plug or a guillotine break in each of these common instrument lines. Examine the effects of erroneous signals on each instrument and on each function (e.g., scrams, trips, permissive signals) that could be actuated or rendered inoperative.

(4) Examine the interactive effects among all systems affected by the common sensing line or sensor failures and the consequential combined effects on the critical reactor parameters.

(5) Compare the consequences of these postulated events with those analyzed in Chapter 15 to ensure the consequences of the postulated events are bounded by the results of the Chapter 15 events and to ensure the postulated events will not require actions or responses beyond the capabilities of the operators or the safety systems. Perform additional transient calculations or analyses necessary to ensure that the worst-case limiting event is bounded by those analyzed in Chapter 15 with the assumption that there is a single active failure in a safety system required to mitigate the effects of the event.

(6) Document the results of the analyses of common sensor and sensor line failures and provide recommendations as appropriate.

The applicant provided the results of this analysis by letter dated August 24, 1984. On the basis of the review of the applicant's analysis, the staff concludes that the effects of control system failures resulting from failure of a power source, sensor, or instrument sensing line are bounded by the Hope Creek FSAR Chapter 15 analyses (i.e., the previously reported limits of minimum critical power ratio, peak reactor vessel and main steamline pressures, and peak fuel cladding temperature for expected operational occurrences would not be exceeded). This resolves Confirmatory Issue 12 as listed in Section 7.1.4.2 of the Hope Creek SER.

It should be noted that the staff is currently reviewing the effects of control system failures at nuclear power plants under Unresolved Safety Issue (USI) A-47, "Safety Implications of Control Systems." In its preliminary conclusions on the resolution of USI A-47, the staff has not identified any significant concerns for BWRs resulting from power source, sensor, or instrument sensing line failures.

7.7.2.2 Credit for Non-Safety-Related Systems in Chapter 15 of the FSAR

During the operating license review of FSAR Sections 7 and 15, the staff assumed that the instrumentation and controls associated with the relief function of the safety/relief valves (SRVs) were safety related. However, the applicant had indicated, in response to a question regarding the use of non-safety-related equipment taken credit for in the FSAR Chapter 15 analysis, that the instrumentation and control equipment associated with the SRV function is not safety related. This did not appear to be consistent with the design of previously reviewed BWRs. This would mean that the control circuits for the SRVs would contain both safety-related circuitry (automatic depressurization system function) and non-safety-related circuitry (relief function) without proper isolation.

As a confirmatory issue, the staff proceeded to review the information provided regarding the relief function of the SRVs and stated that a determination on the adequacy of the design would be provided in a supplement to the Hope Creek SER. By letter dated February 15, 1985, the applicant clarified the response to the staff's question noted above.

The applicant stated that the safety-related relief function of all 14 SRVs is provided by the entirely mechanical, self-actuating action inherent in each of the valves. Pressure relief can also be provided manually by the reactor operator via solenoids actuated by separate remote-manual switches. This manual relief function provides operational flexibility and is not considered a safety function. The Hope Creek design does, however, use safety-grade devices and Class 1E power to perform the manual relief function.

The applicant further stated that the reference provided in the original response to the staff's concern was a reference to the non-safety-related function of the manual relief mode and not to the qualification of the electrical components. Because the components and power supplies of both the manual relief function and the automatic depressurization function are safety grade, no isolation problems exist between these functions. A revised FSAR Table 440.33-1 was provided to clarify this item.

On the basis of the results of its review, the staff has determined that the applicant has provided sufficient information to enable the staff to confirm the adequacy of the design of the relief function of the SRVs. Therefore, Confirmatory Issue 13 as listed in Section 7.1.4.2 of the Hope Creek SER is resolved.

8 ELECTRIC POWER SYSTEMS

8.3 Onsite Power Systems

8.3.3 Common Electrical Features and Requirements

8.3.3.1 Compliance With GDC 2 and 4

8.3.3.1.4 Commitment To Protect All Class 1E Equipment From External Hazards Versus Only Class 1E Equipment in One Division

In Section 8.3.3.1.4 of the SER, the staff documented the applicant's commitment to perform an analysis to verify that Hope Creek could be shut down safely after a main steam tunnel flooding event.

The applicant submitted the main steam tunnel flooding analysis by letter dated May 24, 1985. This analysis identifies all Class 1E equipment and components in the main steam tunnel, Room 4316, that will be subject to the worst-case submergence that results from a break in a main feedwater line. (Flood level is elevation 126 ft of this room.) Also, this report analyzes whether the equipment or component is qualified for submergence. If not qualified, a determination is made whether the equipment or component circuitry has primary and backup protective devices located in a hazard-free area. The purpose of this analysis is to demonstrate that the plant can be safely shut down after both the primary and backup protective device open as a result of the failure of unprotected equipment or component together with the worst-case single failure.

The results of the analysis show that none of components that are flooded and are not qualified for submergence are required for safe shutdown of the plant, nor will their failure prevent safe shutdown. Because of the redundancy of the equipment/systems that are required to safely shut down the plant, no single failure can prevent safe shutdown. On the basis of its evaluation of this report, the staff finds that the analysis satisfies its concerns and, therefore, this confirmatory item is acceptably resolved.

8.3.3.3 Physical Independence - Compliance With GDC 17

8.3.3.3.2 Use of 18 In. Instead of 36 In. of Separation Between Raceways

In Section 8.3.3.3.2 of the SER, the staff was concerned that testing did not substantiate the design of 18 in. instead of 36 in. of vertical separation between redundant cable trays in the cable spreading area, control equipment room, relay room, and main control room.

Subsequently, the applicant submitted test configuration and results to demonstrate that 18-in. separation is adequate. This test configuration consists of two horizontal cable trays with 12 in. of vertical separation between the trays. Both trays were 50% filled with various control and instrumentation cables. The fault cable was a 2/C No. 2 AWG cable located in the top center of the lower tray. Four target cables were located in the bottom center of the upper tray as the worst configuration condition. The purpose of the above configuration test was to demonstrate the adequacy of the design when an electrical fault occurs in the lower tray.

The test results show that the target cables met the acceptance criteria, which are acceptable performance with regard to the insulation resistance test, high potential test, cable continuity test, cable qualification temperature test, and tolerance. On the basis of its evaluation of the test configuration and results, the staff concludes that the 18-in. and the 12-in. separation are adequate for the above described design configuration and, therefore, this confirmatory item is acceptably resolved.

8.3.3.3.3 Specified Separation of Raceway by Analysis and Test

In Section 8.3.3.3.3 of the SER, the staff identified as open items the applicant's justifications for the minimum separation between non-Class 1E conduit and Class 1E cable tray, and the minimum separation between metal-clad cable and Class 1E raceways.

In response to the open items, the applicant submitted a Wyle test report on cable and raceway physical separation verification for Hope Creek and, by letter dated August 5, 1985, provided further information related to tests and analysis for the cable separation open items identified in Paragraphs (2) and (3) of SER Section 8.3.3.3.3.

(2) <u>Non-Class 1E Conduit Separation From Class 1E Tray by a Minimum of 1 In.</u>

The test configuration consisted of a horizontal cable parallel to a rigid conduit with 1-in. vertical separation. The purpose of this test was to demonstrate the adequacy of this design when an electrical fault occurs in a cable in the conduit. The worst-case configuration is represented by a 480-V ac power cable enclosed in a conduit and separated from an open cable tray by 1 in. This power cable is typically a motor control center (MCC) feeder cable with 500 MCM as the largest cable. To provide conservative design values, the fault current available at a unit substation bus is shown, and the duration is based on the operating time of a unit substation circuit breaker. When fault current flows through a breaker, magnetic forces proportional to the square of the instantaneous current are created which tend to blow the contacts apart or otherwise cause mechanical damage. These considerations are most important in establishing the momentary rating of a circuit breaker. In addition to the mechanical stress, the interrupting device must be able to deal with the heat being generated within the arc. It is the root mean square (RMS) value or heating effect that is most important in these tests. Without tripping the breakers by magnetic force, the RMS value of the fault current was used for tested cables to generate maximum thermal energy. This test value is much higher than the overcurrent trip setpoint of the upstream circuit breaker. The test current in this configuration through the 1/C 500 MCM fault cable was 2200 amperes. The total overcurrent test duration was approximately 223 min. The test results show that the target cables met the acceptance criteria, which are acceptable performance with regard to the insulation resistance test, high potential test, cable continuity test, and cable temperature.

On the basis of its evaluation of this test configuration, the fault current baseline, the cable acceptance criteria, and the test results, the staff concludes that a minimum of 1-in. separation between the non-Class 1E conduit and the Class 1E tray is acceptable.

(3) <u>Metal-Clad Cable Separated From Class 1E Raceways by a Minimum of 1 In.</u>

The test configuration consisted of three dropout cables parallel to free-air, metal-clad cable with 1-in. horizontal separation. The purpose of this test was to demonstrate the adequacy of this design when an electrical fault occurs in metal-clad cable in the free air. The test results show that the target cables met the acceptance criteria for the cable.

On the basis of its evaluation of the test configuration and results, the staff concludes that 1-in. horizontal separation between the metal-clad cable and Class 1E raceways is adequate and is, therefore, acceptable.

8.3.3.3.4 Use of an Inverter as an Isolation Device

In Section 8.3.3.3.4 of the SER, the staff identified a confirmatory item regarding the use of an inverter as an isolation device. By letter dated March 7, 1985, the applicant submitted a copy of "Test Report for a 20 KVA UPS System Power Circuit Isolation Test," prepared for Hope Creek. The intent of the test program was to show that the uninterruptible power supply (UPS)/inverter qualifies as a power circuit isolation device between Class 1E power input sources and non-Class 1E load circuits, as defined by IEEE Std. 384-1981 and RG 1.75.

The staff concluded that the proposed testing of the isolation device should include the four fault conditions listed in the test plan as well as a 200% overload to demonstrate that the input current and voltage do not exceed specified values. Subsequently, the applicant has excluded the hot short test. The basis for this exclusion is that design changes were implemented to ensure that none of the inverter's output cables are routed in raceways containing any 480-V ac service level cables. In addition, the 200% overload test in the test program could not be performed because of design limitations of the inverter. The inverter's current limiting feature would not support a 200% overload. Application of 200%, or any load above the inverter's current limit setting, will cause the static switch to transfer the load (overload) to the backup ac source. The transfer to the backup source will be annunciated by the UPS. On the basis of its evaluation of the above two exclusions from the previously approved test plan, the staff finds the justification acceptable.

The test program was designed to measure the effects on the inputs to the UPS when its output is grounded or shorted (faulted). During a shorted or grounded UPS output fault condition, the UPS system must function as both power circuit isolation device and as an input current limiter to limit the effects of the faults on the Class 1E power sources to acceptable values.

The test program and its acceptance criteria were described in the SER. The test results show that the input sources (alternating dc, normal ac, and backup ac) were not affected by connecting an output line ground. There is no disturbance indicated on any input source wave form, at or after the ground fault application in all UPS configurations. The alternate dc and normal ac sources were not adversely affected by applying any faults to either the inverter or

UPS outputs. The maximum fault current requirement from the "BACK-UP" is substantially sinusoidal 60-Hz current for approximately 1.5 sec until the fuse melts out, but the Hope Creek MCC circuit breaker will clear the fault in less than 1.5 sec.

On the basis of its assessment of the test results, the staff concludes that the applicant has successfully demonstrated that the UPS/inverter is an adequate isolation device as used in this application. This item is, therefore, resolved.

9 AUXILIARY SYSTEMS

9.3 Process Auxiliaries

9.3.2 Process and Postaccident Sampling Systems

In the SER, the staff concluded that the Hope Creek postaccident sampling system meets the requirements of Item II.B.3 of NUREG-0737 and is, therefore, acceptable. However, as a confirmatory issue, the staff noted that the applicant should provide a plant-specific procedure to estimate the extent of core damage after an accident, pursuant to Criterion 2 of Item II.B.3. Additionally, in the SER, the staff stated that it would condition the license to specify that the postacci-dent sampling system should be operational before 5% power is exceeded.

By letter dated June 24, 1985, the applicant provided a plant-specific procedure for estimating core damage during accident conditions based on the generic BWR Owners Group core-damage assessment methodology dated June 17, 1983. Core-damage estimates are based on utilizing postaccident sampling system measurements of fission product concentrations in the primary coolant and in the containment. Additional procedures are provided for estimating the extent of metal-water reaction on the basis of measured hydrogen concentration in the containment and for estimating the extent of core damage on the basis of containment radiation monitors. Reactor vessel water level is used to establish if there has been adequate core cooling. This procedure meets Criterion 2 of Item II.B.3 and is, therefore, acceptable.

On the basis of the above evaluation, the staff concludes that the applicant's postaccident sampling system meets all the requirements of Item II.B.3 of NUREG-0737 and is, therefore, acceptable. The staff considers SER Confirmatory Issue 31 resolved and removes proposed SER License Condition 4.

12 RADIATION PROTECTION

12.3 Radiation Protection Design Features

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

Section II.4.b of SRP Section 12.3-12.4 states that the air monitoring system "should be capable of detecting ten MPC (maximum permissible concentration) hours of particulate and iodine radioactivity from any compartment which has a possibility of containing airborne radioactivity and which normally may be occupied by personnel." At the time the SER was issued, the applicant could not provide the staff with the locations, quantity, and type of continuous airborne (radiation) monitors that would be used at Hope Creek. For this reason, airborne iodine concentration instrumentation was identified as a confirmatory issue.

By letter dated July 1, 1985, the applicant stated that to comply with the above guidance, he would position 15 portable continuous air monitors (CAMs) at various locations (including the radwaste control room area, access control area, and spent fuel pool area) within the station during normal operations. These monitors may be augmented or shifted, as needed, during outages or special evaluations. The applicant will use these CAMs to provide in-plant monitoring of particulates and iodine levels. These monitors will be capable of detecting 10 MPC hours of particulate and iodine radioactivity and will be calibrated at 6-month intervals. The applicant will use these CAMs, in conjunction with in-plant surveys, to prevent exposure of personnel to high concentrations of airborne activity in radiation areas. The locations, quantity, and type of CAMs as described by the applicant in the letter of July 1, 1985, are acceptable for use at Hope Creek. On the basis of the applicant's submission of this information and his commitment to revise the FSAR (in Amendment 12) to include the information contained in the July 1, 1985, letter, Confirmatory Issue 32 is considered resolved.

12.5 Operational Radiation Protection Program

12.5.2 Facilities, Equipment, and Instrumentation

In the SER, the staff stated that information on onsite instrumentation was confirmatory pending the applicant's submittal of this information. By letter dated July 1, 1985, the applicant submitted the necessary information.

In this letter the applicant stated that he will be able to perform onsite electronic and radiation calibrations of dose rate and count rate instruments, counting scalars, and portal monitors. Radiation sources used for calibration will include gamma, beta, alpha, and neutron sources. Various low-activity-exempt check sources will also be used to verify instrument response.

All instruments will be calibrated in compliance with the American National Standards Institute (ANSI) N323-1978, which establishes calibration methods for portable radiation protection instruments used for the detection and measurement of ionizing radiation and radioactive surface contamination. On the basis of the applicant's submission of this information and his commitment to revise the FSAR (in Amendment 12) to include the information contained in the July 1, 1985, letter, Confirmatory Issue 34 is considered resolved.

Section III.D.3.3 of NUREG-0737 states that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. Because this information was not available for staff review when the SER was published, this was identified as a confirmatory issue. By letter dated June 18, 1985, the applicant provided the required information. This information was incorporated in the FSAR by Amendment 11.

The applicant states that he will use two types of portable instruments for in-plant sampling for radioiodine. These are a standard low-volume air sampler with a silver zeolite cartridge as the sample medium, and an emergency air sampler assembly, consisting of an evacuated Marinelli beaker with a silver zeolite cartridge as the sample medium. In addition to the ac-powered low-volume air samplers used for normal onsite air sampling, two dc-powered air samplers will be provided in each of the emergency lockers located in the 137-ft elevation control point, technical support center, control room, and emergency van, and one in each offsite survey team kit.

Before analysis, the applicant will purge all silver zeolite cartridges analyzed in plant using bottled nitrogen gas or clean air to ensure the absence of noble gases. This purging will be performed in a well-ventilated area or under a laboratory hood. The inplant and onsite silver zeolite cartridges will then be analyzed using high-purity germanium detectors, which are located in the chemistry laboratory and the radiation protection count room. Offsite silver zeolite cartridges will be analyzed using an Eberline Model SAM-2 portable dual-channel analyzer with a probe capable of detecting 365 kev iodine-131. The SAM-2 can also be used to analyze inplant and onsite silver zeolite cartridges if background radiation levels in the chemistry laboratory or radiation protection count room are too high to perform analyses with the high-purity germanium detectors. As suggested in NUREG-0737, the applicant has developed procedures and training necessary to ensure proper use of this equipment during normal and accident conditions. The training consists of both classroom instruction and practical factors demonstrations.

On the basis that the applicant has adequately addressed the criteria of Item III.D.3.3 of NUREG-0737, Confirmatory Issue 35 is considered resolved.

15 SAFETY ANALYSIS

15.7 Radioactive Releases From a Subsystem or Component

15.7.5 Spent Fuel Cask Drop Accidents

Section 15.7.5 of the SER stated that the need for computing the radiological consequences of a spent fuel cask drop accident had not been determined and that the license would be conditioned to require resolution of this issue before a spent fuel cask is moved within the plant.

The staff has reviewed additional information supplied by the applicant in the resolution of SER Outstanding Issue 8, "Control of Heavy Loads." The spent fuel cask is equipped with redundant sets of lifting lugs and yokes compatible with the single-failure-proof reactor building crane, thus preventing a cask drop caused by a single failure. In addition, in Section 9.1.5 of Supplement No. 1 to the SER, the staff has concluded that the overhead heavy load handling systems were adequately designed to meet the guidelines of NUREG-0612. On the basis of these findings and on compliance with SRP Section 15.7.5, no radiological consequences of a cask drop accident have been computed. The staff concludes that the spent fuel cask is adequately protected against drop accidents and removes SER License Condition 7.

APPENDIX A

CONTINUATION OF CHRONOLOGY

January 30, 1979	Letter from applicant regarding reactor coolant pressure boundary (RCPB).
November 4, 1981	Letter to applicant regarding RCPB.
February 10, 1983	Letter from applicant regarding RCPB.
June 13, 1985	Letter from applicant forwarding marked-up revisions to Final Safety Analysis Report (FSAR), per NUREG-0612 regarding control of heavy loads and SER Outstanding Issue 8. Changes reflect revised load paths for certain hoists and deletion of others. Includes updated design information.
June 18, 1985	Letter from applicant forwarding response to SER Confirmatory Issue 35 concerning airborne iodine concentration instruments. Response describes equipment, training, and procedures for determining iodine concentration. FSAR will be revised.
June 21, 1985	Letter from applicant forwarding status of Technical Specification issues identified in SER Section 16. Resolution of listed Technical Specification issues is considered incomplete.
June 24, 1985	Letter from applicant forwarding "Estimation of Reactor Core Damage Under Accident Conditions," per SER Confirmatory Issue 31. Information will be incorporated into next FSAR amendment.
June 25, 1985	Letter from applicant forwarding index of environmental equipment summary sheets (EESSs), revised FSAR Table 3.11-5, revised EESS, hazards walkdown status summary, and example of average temperature for life determination.
June 25, 1985	Letter from applicant responding to Pump and Valve Operability Review Team (PVORT) SER Confirmatory Issue B.1 concerning location of high pressure coolant injection turbine nameplate. Revised schedule of remaining audit issues was enclosed.
June 27, 1985	Letter from applicant forwarding response to Generic Letter 85-07 regarding implementation of integrated schedules for plant modifications. Integrated living schedule is being implemented for internal use.

June 28, 1985	Generic Letter 85-11 to all licensees of operating reactors concerning completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612.
June 28, 1985	Generic Letter 85-12 to applicants and licensees with Westinghouse-designed nuclear steam supply systems regarding implementation of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps."
June 28, 1985	Letter from applicant forwarding Revision 1 to "Environmental Equipment Summary Sheets (EESS) Index." Revision supersedes EESS index submitted on June 25, 1985. EESS 153 and 155 and attachment to EESS 73 were also enclosed for incorporation into June 25, 1985, submittal.
July 1, 1985	Letter from applicant forwarding status of SER Sections 1.7 and 1.8 open and confirmatory items and resolution of items that will be incorporated into FSAR Amendment 12.
July 2, 1985	Letter from applicant requesting permission to adopt use of later editions of ASME Code, Section III requirements to complete as-built piping reconciliation in timely manner.
July 2, 1985	Letter from applicant submitting additional information requested per May 29, 1985, telcon regarding plant-unique analysis report hydrodynamic load question responses previously submitted in January 31, 1985, letter. Ring beam frequencies were calculated using improved Rayleigh method.
July 2, 1985	Letter to applicant advising that applicant has not fully complied with listed commitments regarding May 9, 1985, request to use ASME Code, Case N-411, "Alternate Damping Values for Seismic Analysis...." Use of case acceptable when listed actions are performed.
July 3, 1985	Letter from applicant providing justification to support request for approval to eliminate postulation of intermediate pipe breaks as specified by Sections 3.6.2.II.1 and II.2 of the SRP.
July 3, 1985	Letter to applicant forwarding first draft of Technical Specifications based on applicant's January 17, February 7, March 22, April 10, and June 10, 1985, submittals and General Electric Standard Technical Specifications - BWR/4.
July 7, 1985	Letter to applicant responding to June 7, 1985, request for authorization to use ASME Code, Case N-413, "Minimum Size of Fillet Welds for Linear Type Supports..." at facility. Code case acceptable and should be documented in future FSAR amendment.

July 10, 1985	Letter from applicant summarizing July 2, 1985, telcon concerning staff's July 1, 1985, verbal request for additional information on May 23, 1985, amended special nuclear materials license application. Summary includes description of training for persons involved in fuel-handling process.
July 12, 1985	Letter from applicant forwarding Revision 0 to "Pump and Valve Inservice Testing Program" and proprietary Drawing 11872210, "Starting and Control Air System," concerning SER Confirmatory Issue 3.
July 15, 1985	Letter from applicant forwarding preliminary Attachments 3-5 to Procedure OP-AP-ZZ-101(Q), Revision 1 to Procedure SA-AP-ZZ-002(Q), and Revision 2 to Procedure SA-AP-ZZ-004(Q), in response to staff's request for additional information regarding Generic Letter 83-28.
July 15, 1985	Letter from applicant forwarding FSAR Amendment 11. Description of revisions by FSAR section is enclosed.
July 16, 1985	Letter from applicant forwarding response to PVORT Open Item 1.A from PVORT/Seismic Qualification Review Team (SQRT) May 7-10, 1985, audit regarding determination of cause of diminished seal water leakoff flow experienced by service water pump AD-502.
July 17, 1985	Letter to applicant forwarding results of review and evaluation of adequacy of emergency plan through Revision 7 and commitments in May 30, 1985, letter.
July 19, 1985	Letter from applicant forwarding status of open and confirmatory items identified in SER Sections 1.7 and 1.8. Resolution of Confirmatory Issues 6 and 12 also enclosed for review and approval. Resolutions will be incorporated into FSAR Amendment 12.
July 19, 1985	Letter from applicant notifying staff of change in schedule for implementation of required station procedures. All procedures required for station operation will be in place at least 90 days before fuel load. FSAR Amendment 12 will be revised.
July 19, 1985	Letter from applicant forwarding July 19, 1985, affidavit certifying distribution of FSAR Amendment 11 per 10 CFR 2.101.
July 22, 1985	Letter from applicant notifying staff of August 1, 1985, meeting with NRC and technical personnel to discuss accelerated power ascension test program.

July 23, 1985	Letter from applicant forwarding Revision 4 to security plan and Revision 1 to security contingency plan. Revisions withheld per 10 CFR 73.21.
July 25, 1985	Letter from applicant forwarding "Report of Resolution of Electromagnetic Interference Effects on Bailey Input Logic Modules," submitted in response to SER Outstanding Issue 5 regarding solid-state logic modules.
July 25, 1985	Letter from applicant confirming July 15, 1985, telcon that applicant is considering staff's December 24, 1984, recommendations regarding "Evaluation of Model for Predicting Drywell to Wetwell Vacuum Breaker Valve Dynamics."
July 25, 1985	Letter from applicant responding to staff's March 1, 1985, request for additional information on preservice inspection of pipe welds with corrosion-resistant cladding. Actions since November 26, 1984, progress report and Southwest Research Institute field inspection procedure are enclosed.
July 25, 1985	Letter to applicant confirming safety parameter display system (SPDS) design verification and design validation audit on August 27-28, 1985. Staff audit plan is enclosed for use in preparing for audit.
July 26, 1985	Letter from applicant forwarding response to SER Confirmatory Issue 1 regarding feedwater isolation check valve analysis. Information will be incorporated into FSAR Amendment 12.
July 26, 1985	Letter to applicant forwarding draft environmental protection plan (nonradiological). Review is requested.
July 29, 1985	Letter from applicant forwarding revised procedures generation package (PGP), per May 1, 1985, request for additional information on SER Outstanding Issue 14. PGP will be incorporated into FSAR Amendment 12. Response to items regarding specific technical guidelines is also enclosed.
July 29, 1985	Letter from applicant forwarding, for review, mechanical equipment qualification audit on July 15-18, 1985.
July 29, 1985	Letter to applicant advising that staff will perform electrical audit on August 7-8, 1985, which will consist of tour of facility and meetings to resolve all outstanding SER open and confirmatory issues in electrical power systems area.
July 31, 1985	Letter from applicant submitting information on equipment qualification audit on July 15-18, 1985, including revised environmental equipment summary sheet, Pyco temperature element and Anaconda flex conduit justifications, equipment traceability, position on maximum service temperature, and solenoid valve information.

July 31, 1985	Letter from applicant requesting permission to adopt provisions of Revision 2 to "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants." Revised FSAR pages are enclosed.
July 31, 1985	Letter from applicant responding to staff's May 14, 1985, oral request for additional information on SPDS isolation.
August 1, 1985	Generic Letter 85-14 to all licensees regarding commercial storage and power reactor sites of low-level radwaste not generated by utility.
August 5, 1985	Generic Letter 85-13 to all reactor licensees and applicants transmitting NUREG-1154 regarding Davis-Besse loss of main and auxiliary feedwater event.
August 5, 1985	Letter from applicant forwarding additional information on test currents versus maximum fault currents and test performance of metal-clad cable compared with rigid steel conduit, per May 25, 1985, telcon regarding SER Open Issue 7.
August 5, 1985	Letter from applicant forwarding list of resolutions for FSAR commitments for June and July 1985.
August 5, 1985	Letter from applicant forwarding response to PVORT Open Item 1.b concerning service water pump AS-502 functionality, per PVORT/SQRT May 7-10, 1985, audit.
August 5, 1985	Letter to applicant forwarding marked-up draft Technical Specifications, current as of August 5, 1985. Working meeting will be held on August 12, 1985, on site to resolve differences.
August 6, 1985	Generic Letter 85-15 to all licensees of operating reactors regarding information on deadlines for compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
August 6, 1985	Letter from applicant requesting permission to purchase Gorman-Rupp Co replacement parts for eight Crane Deming fuel oil and lube oil pumps. Crane Deming sold pump line to Gorman-Rupp. Parts would meet all ASME Code, Section III requirements with exception of "N" stamp.
August 7, 1985	Letter from applicant forwarding preliminary Attachments 1 and 2 to Revision B to Operating Procedure OP-AP-ZZ-101(Q), "Post-Reactor Scram/ECCS Actuation Review and Approval Requirements," for review.
August 7, 1985	Summary of July 31, 1985, meeting with representatives from the applicant, Sargent & Lundy, and State of New Jersey in Bethesda, Maryland, regarding independent design verification program (IDVP) summary report.

August 9, 1985	Letter from applicant forwarding information on steam/water hammer effects to support applicant's request for approval to eliminate postulation of intermediate pipe breaks (SRP Sections 3.6.2.II.1 and II.2), in response to staff's request.
August 9, 1985	Letter from applicant forwarding Revision 8 to Emergency Plan Procedure M1-7, "Decontamination and Treatment of Radioactively Contaminated Patient at Salem County Memorial Hospital," including response to commitments made at May 16 and 17, 1985, meetings.
August 12, 1985	Letter from applicant requesting approval of proposed change to SER Section 9.5.1.3, revising fire brigade staffing levels to six-man dedicated brigade. Proposal is based on applicant's reorganization and reassignment of job duties.
August 12, 1985	Letter to applicant forwarding Supplement 2 to SER regarding application for operating license.
August 15, 1985	Letter from applicant forwarding, for review, information on Items 4-6 of July 15-18, 1985, equipment qualification audit, revised response to Audit Item 3 regarding use of Anaconda flex conduit-drywell, and Revision 2 to "Environmental Qualification Summary Report for Hope Creek Generating...."
August 16, 1985	Letter from applicant forwarding responses to SQRT Confirmatory Issues 1, 2, 4, and 6 and SQRT Generic Open Issue 1, per NRC PVORT/SQRT May 5-10, 1985, audit.
August 19, 1985	Summary of August 1, 1985, meeting with representatives from the applicant, General Electric, State of New Jersey, and Conner & Wetterhahn in Bethesda, Maryland, regarding acceleration of power ascension program.
August 21, 1985	Letter from applicant forwarding Revision 0 to "Process Control Program," which supersedes April 8, 1985, submittal. Program addresses areas of concern discussed during June 19, 1985, meeting and identified in NRC May 15, 1985, draft, "Guidelines for...Process Control Program."
August 21, 1985	Letter from applicant forwarding safety evaluations of 5 of 29 power ascension test modifications, including Test 17, "Core Performance," Test 19, "Core Power-Void Mode Response," and Test 25, "Turbine Trip and Generator Load Rejection."
August 23, 1985	Generic Letter 85-16 to all licensees of operating reactors and applicants for operating licenses regarding high boron concentrations.
August 23, 1985	Generic Letter 85-17 to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits regarding availability of Supplements 2 and 3 to NUREG-0933, "Prioritization of Generic Safety Issues."

August 23, 1985	Letter to applicant forwarding agenda for site visit scheduled for September 16-19, 1985. Visit will concentrate on areas to aid in resolution of SER open and confirmatory issues, including physical separation between safety-related circuits.
August 26, 1985	Letter from applicant forwarding revised response to FSAR Question 421.42 regarding failure of reactor controls and vital instruments to reflect re-review of Office of Inspection and Enforcement (IE) Circular 79-02 as required by IE Bulletin 79-27. Item closes out SER Confirmatory Issue 22.
August 26, 1985	Letter from applicant responding to NRC July 2, 1985, request for additional information on applicant's request for authorization to use ASME Code, Case N-411, and selected portions of 1981 Winter Addenda and 1973 Edition of Section III of ASME Code.
August 26, 1985	Letter to applicant informing that June 13, 1985, changes to FSAR Amendment 11 concerning revised and deleted load paths for hoists are in compliance with Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."
August 26, 1985	Letter from applicant forwarding State of New Jersey Pollutant Discharge Elimination System Permit NJ0025411, per Section 1.3 of Final Environmental Statement (NUREG-1074).
August 27, 1985	Letter from applicant forwarding responses to 14 unresolved items identified in Inspection Report 50-354/85-24. Revised, marked-up FSAR text, closing unresolved fire protection items, also was enclosed. Revised text will be included in FSAR Amendment 12.
August 28, 1985	Letter from applicant submitting schedule for expected submittals on modifications to power ascension test program. Justification states enclosed Test 30 will be addressed in applicant's response to Generic Letter 84-23.
August 28, 1985	Letter to applicant forwarding Generic Letter 85-15 concerning deadlines for compliance with 10 CFR 50.49.
September 3, 1985	Letter from applicant forwarding revised August 5, 1985, response to Outstanding Issue 7 regarding actual fault current values, per staff's August 7-8, 1985, audit.
September 4, 1985	Letter from applicant submitting additional information on SER Confirmatory Issue 1 concerning stresses imposed on feedwater check valve hinge following double-ended break of feedwater line outside containment.
September 4, 1985	Letter to applicant forwarding request for additional information to address Advisory Committee on Reactor Safeguards December 18, 1984, concern regarding control room habitability in event of loss of both emergency ventilation trains.

September 4, 1985	Letter to applicant forwarding second draft of Technical Specifications for review and identification of statements inaccurately reflecting FSAR or as-built plant.
September 9, 1985	Letter from applicant requesting permission to procure replacement parts for ASME Code, Section III components without "N" stamp in cases where manufacturer is no longer stamp holder.
September 9, 1985	Letter from applicant clarifying SER Confirmatory Issue 1 regarding stresses imposed on feedwater check valve hinge pin following double-ended break of feedwater line outside containment.
September 9, 1985	Letter from applicant forwarding Sargent & Lundy August 30, 1985, letter transmitting Vols. 1-7 of "Hope Creek Generating Station IDVP," final report dated August 30, 1985.
September 10, 1985	Letter from applicant informing staff that General Atomic Co contracted to provide radiation monitoring system for facility since Technology for Energy Corp filed for reorganization under Chapter 11 of Bankruptcy Act on March 29, 1985.

APPENDIX B

BIBLIOGRAPHY

Continuum Dynamic, Inc., Generic Report CDI 84-3, "Mark I Wetwell to Drywell Vacuum Breakers."

Dente, J., BWR Owners Group, letter to D. Eisenhut, NRC, "Transmittal of Generic Procedures for Estimation of Core Damage Using Post Accident Sampling Systems," June 17, 1983.

U.S. Nuclear Regulatory Commission, Generic Letter 83-08, Subject: Modification of Vacuum Breakers on Mark I Continments, Feb. 2, 1983.

---, NUREG-0484, "Methodology for Combining Dynamic Responses," Rev. 1, May 1980.

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

---, NUREG-0661, "Safety Evaluation Report Mark I Containment Long-Term Program," July 1980; Supplement No. 1, Aug. 1982.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (includes branch technical positions).

---, NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," Aug. 1981.

---, Office of Inspection and Enforcement, IE Bulletin 79-15, "Deep Draft Pump Deficiencies," July 11, 1979.

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

---, IE Circular 79-02, "Failure of 120-V Vital AC Power Supplies," Jan. 16, 1979.

---, IE Information Notice 79-22, "Qualification of Control Systems," Sept. 17, 1979.

---, IE Information Notice 83-42, "Reactor Mode Switch Modifications," June 23, 1983.

Vassalo, D., NRC, letter to H. Pfefferlen, GE, "Mark I Wetwell to Drywell Vacuum Breakers," Dec. 24, 1984.

APPENDIX D

ACRONYMS AND INITIALISMS

ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	branch technical position
BWR	boiling-water reactor
CAM	continuous air monitor
CDI	Continuum Dynamic, Inc.
CFR	*Code of Federal Regulations*
CRC	corrosion-resistant cladding
DCP	design change package
DOE	U.S. Department of Energy
ECC	emergency core cooling
EESS	environmental equipment summary sheet
EOC	end of cycle
FSAR	Final Safety Analysis Report
GDC	general design criterion(a)
GE	General Electric
HCU	hydraulic control unit
HELB	high energy line break
HPCI	high pressure coolant injection
ID	inside diameter
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
ISI	inservice inspection
LOCA	loss-of-coolant accident
MCC	motor control center
MG	motor generator
MPC	maximum permissible concentration
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OD	outside diameter

PGP	procedures generation package
PSE&G	Public Service Electric and Gas Company
PSI	preservice inspection
PUAR	plant-unique analysis report
PVORT	Pump and Valve Operability Review Team
RACS	reactor auxiliaries cooling system
RCPB	reactor coolant pressure boundary
RG	regulatory guide
RHR	residual heat removal
RMS	root mean square
RPT	recirculation pump trip
RPV	reactor pressure vessel
RRS	reactor recirculation system
SACS	safety auxiliaries cooling system
SDV	scram discharge volume
SER	Safety Evaluation Report
SLC	standby liquid control
SPDS	safety parameter display system
SQRT	Seismic Qualification Review Team
SRP	Standard Review Plan
SRV	safety/relief valve
TACS	turbine auxiliaries cooling system
TMI	Three Mile Island
UPS	uninterruptible power supply
USI	unresolved safety issue

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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Name	Title	Branch
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B. Brown	Idaho National Engineering Laboratory
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APPENDIX N

TECHNICAL EVALUATION OF THE HOPE CREEK GENERATING STATION
PLANT-UNIQUE ANALYSIS REPORT

TECHNICAL EVALUATION OF THE HOPE CREEK GENERATING STATION

PLANT-UNIQUE ANALYSIS REPORT

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July 1985

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ABSTRACT

This Technical Evaluation Report (TER) presents the results of the post-implementation audit of the Plant Unique Analysis Report (PUAR) for the Hope Creek Generating Station. The contents of the PUAR were compared against the hydrodynamic load Acceptance Criteria (AC) contained in NUREG-0661. The TER summarizes the audit findings (Table 1), and discusses the nature and status of any exceptions to the AC, identified during the audit (Table 2).

ACKNOWLEDGEMENTS

The cognizant NRC Technical Monitor for this program was Dr. Farouk Eltawila of the Containment Systems Branch (DSI) and the NRC Project Manager was Mr. Lawrence Ruth of the Containment Systems Branch.

List of Acronyms

AC	Acceptance Criteria
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CO	Condensation Oscillation
DBA	Design Basis Accident
DSI	Division of Systems Integration
FSAR	Final Safety Analysis Report
FSI	Fluid Structure Interaction
FSTF	Full Scale Test Facility
HCGS	Hope Creek Generating Station
LDR	Load Definition Report
LOCA	Loss-of-Coolant Accident
LTP	Long Term Program
NRC	Nuclear Regulatory Commission
PSE&G	Public Service Electric and Gas Company
PUA	Plant-Unique Analysis
PUAR	Plant-Unique Analysis Report
QSTF	Quarter Scale Test Facility
RFI	Request For Information
SER	Safety Evaluation Report
SMA	Structural Mechanics Associates
SPTMS	Suppression Pool Temperature Monitoring System
S/RV	Safety/Relief Valve
S/RVDL	Safety/Relief Valve Discharge Line
STP	Short Term Program
TAP	Torus Attached Piping
TER	Technical Evaluation Report
T/Q	T-Quencher

1. INTRODUCTION

The suppression pool hydrodynamic loads associated with a postulated loss-of-coolant accident (LOCA) were first identified during large-scale testing of an advanced design pressure-suppression containment (Mark III). These additional loads, which had not explicitly been included in the original Mark I containment design, result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus). Because these hydrodynamic loads had not been considered in the original design of the Mark I containment, a detailed reevaluation of the Mark I containment system was required.

A historical development of the bases for the original Mark I design as well as a summary of the two-part overall program (i.e., Short Term and Long Term Programs) used to resolve these issues can be found in Section 1 of Reference 1. Reference 2 describes the NRC staff's evaluation of the Short Term Program (STP) used to verify that licensed Mark I facilities could continue to operate safely while the Long Term Program (LTP) was being conducted.

The objectives of the LTP were to establish design-basis (conservative) loads that are appropriate for the anticipated life of each Mark I BWR facility (40 years), and to restore the originally intended design-safety margins for each Mark I containment system. The principal thrust of the LTP has been the development of generic methods for the definition of suppression pool hydrodynamic loadings and the associated structural assessment techniques for the Mark I configuration. The generic aspects of the Mark I Owners Group LTP were completed with the submittal of the "Mark I Containment Program Load Definition Report" (Ref. 3) and the "Mark I Containment Program Structural Acceptance Guide" (Ref. 4), as well as supporting reports on the LTP experimental and analytical tasks. The Mark I containment LTP Safety Evaluation Report (NUREG-0661) presented the NRC staff's review of the generic suppression pool hydrodynamic load definition and structural assessment techniques proposed in the reports cited above. It was concluded that the load definition procedures utilized by the Mark I Owners Group, as modified by NRC requirements, provide conservative estimates of these loading conditions and that the structural acceptance criteria are consistent with the requirements of the applicable codes and standards.

The generic analysis techniques are intended to be used to perform a plant-unique analysis (PUA) for each Mark I facility to verify compliance with the acceptance criteria (AC) of Appendix A to NUREG-0661. The objective of this study is to perform a post-implementation audit of the Hope Creek plant-unique analysis (Reference 5) against the hydrodynamic load criteria in NUREG-0661.

2. POST-IMPLEMENTATION AUDIT SUMMARY

The purpose of the post-implementation audit was to evaluate the hydrodynamic loading methodologies which were used as the basis for modifying the pressure suppression system of the Hope Creek Generating Station. The Hope Creek PUAR methodologies (Reference 5) were compared with those of the LDR (Reference 3) as approved in the AC of NUREG-0661 (Reference 1). The audit procedure consisted of a moderately detailed review of the plant unique analysis report (PUAR) to verify both its completeness and its compliance with the acceptance criteria. A list of requests for further information was submitted (Reference 6), and a written response was obtained from the licensee (Reference 7). Additional clarification was obtained during two teleconferences held on May 19, 1985 and July 22, 1985, and from a brief letter from PSE&G to NRC dated July 2, 1985.

Table 1 summarizes the audit results. It lists the various load categories specified in the AC, and indicates plant-unique information through the references, in the right-hand column, to the notes which follow in the text.

LOADS	NUREG-0661 AC SECTION	CRITERIA		NOT APPLICABLE	ALTERNATE APPROACH	NOTES
		MET	NOT MET			
CONTAINMENT PRESSURE & TEMPERATURE	2.1	✓				
VENT SYSTEM THRUST LOADS	2.2	✓				
<u>POOL SWELL</u>						
TORUS NET VERTICAL LOADS	2.3	✓				
TORUS SHELL PRESSURE HISTORIES	2.4	✓				
VENT SYSTEM IMPACT AND DRAG	2.6	✓				
IMPACT AND DRAG ON OTHER STRUCTURES	2.7	✓				
FROTH IMPINGEMENT	2.8	✓				1
POOL FALLBACK	2.9	✓				
LOCA JET	2.14.1				✓	2
LOCA BUBBLE DRAG	2.14.2				✓	2
VENT HEADER DEFLECTOR LOADS	2.10	NA				

TABLE 1. LOAD CHECKLIST FOR POST-IMPLEMENTATION AUDIT

LOADS	NUREG-0661 AC SECTION	CRITERIA		NOT APPLICABLE	ALTERNATE APPROACH	NOTES
		MET	NOT MET			
<u>CONDENSATION OSCILLATION</u>						
TORUS SHELL LOADS	2.11.1				✓	3
LOADS ON SUBMERGED STRUCTURES	2.14.5				✓	2,3,4
VENT SYSTEM LOADS	2.11.3	✓				
DOWNCOMER DYNAMIC LOADS	2.11.2	✓				
<u>CHUGGING</u>						
TORUS SHELL LOADS	2.12.1				✓	3
LOADS ON SUBMERGED STRUCTURES	2.14.6				✓	2,3,4
VENT SYSTEM LOADS	2.12.3	✓				
LATERAL LOADS ON DOWNCOMERS	2.12.2	✓				

TABLE 1. (CONTINUED)

LOADS	NUREG-0661 AC SECTION	CRITERIA MET	CRITERIA NOT MET	NOT APPLICABLE	ALTERNATE APPROACH	NOTES
T-QUENCHER LOADS						
DISCHARGE LINE CLEARING	2.13.2	✓				
TORUS SHELL PRESSURES	2.13.3	✓				5
JET LOADS ON SUBMERGED STRUCTURES	2.14.3	✓				
AIR BUBBLE DRAG	2.14.4				✓	2,6
THRUST LOADS ON T/Q ARMS	2.13.5	✓				
S/RVDL ENVIRONMENTAL TEMPERATURES	2.13.6	✓				

TABLE 1. (CONTINUED)

	DESCRIPTION	NUREG-0661 AC SECTION	CRITERIA MET	CRITERIA NOT MET	NOT APPLICABLE	ALTERNATE APPROACH	NOTES
1	SUPRESSION POOL TEMPERATURE LIMIT	2.13.8	✓				7
2	SUPRESSION POOL TEMPERATURE MONITORING SYSTEM	2.13.9	✓				7
3	DIFFERENTIAL PRESSURE CONTROL SYSTEM FOR THOSE PLANTS USING A DRYWELL-TO-WETWELL PRESSURE DIFFERENCE AS A POOL SWELL MITIGATOR	2.16	NA				
4	SRV LOAD ASSESSMENT BY IN-PLANT TEST	2.13.9	✓				5

TABLE 1. (CONTINUED)

Notes to Table 1

Number	
1	For some structures, Region I froth loads were calculated using the high-speed QSTF movies. This alternative is outlined in Appendix A of the AC.
2	Instead of the equivalent cylinder procedure specified in the AC to calculate acceleration drag volumes on sharp cornered submerged structures, the PUAR selected alternate modeling of the structures and used published acceleration volumes. The discussion in Section 3.1 explains why this procedure was found acceptable.
3	To calculate CO and post-chug loads on the torus shell as well as on submerged structures, the 50 individual load harmonics were combined using a random phasing technique instead of the absolute summation specified in the AC. The discussion of Section 3.2 describes why this alternate method was found acceptable.
4	To account for FSI effects during CO and chugging submerged structure loads, the AC suggested adding torus boundary accelerations directly to local fluid accelerations. Instead, the applicant used a method which calculated FSI acceleration fields anywhere in the torus based on knowing the boundary accelerations. This method, which has been accepted during previous PUAR reviews, is discussed in Section 3.3.

Number

5 The analytical model to calculate SRV torus shell loads approved in the AC was modified slightly before being applied to Hope Creek. The purpose of the modifications was to more closely bound the pressure traces observed in the Monticello tests on which the model is based. These changes have been found acceptable. Future SRV tests will be conducted in the Hope Creek plant to further confirm that the analytically obtained loadings are conservative.

6 For SRV air bubble drag loads, the applicant reduced the AC bubble pressure bounding factor of 2.5 to 1.75. This still bounded peak positive bubble pressure and maximum bubble pressure differential from the Monticello test data. Dynamic load factors were derived from Monticello's in-plant SRV test data. These modifications have been found acceptable and are discussed in Section 3.4.

7 While no pool temperature information is contained in the PUAR, the suppression pool temperature analysis for Hope Creek, along with a description of the suppression pool temperature monitoring system (SPTMS) can be found in section 6.2.1.1.10 of the Hope Creek FSAR. The SPTMS, as well as the pool temperature analysis have been found acceptable.

3. EXCEPTIONS TO GENERIC ACCEPTANCE CRITERIA

Hope Creek is one of several plants analyzed by NUTECH Engineers, Inc. based on an essentially common hydrodynamic loading methodology (Fermi, Duane Arnold, Monticello, Quad Cities and Dresden are other plants in this group). The methodology differs from the generic acceptance criteria of NUREG-0661 in four major areas which are listed in Table 2.

In what follows, each of these areas is discussed in detail, and the bases for the resolutions of the differences indicated.

Table 2: Issues Identified During Audit as Exceptions to the Generic Acceptance Criteria

Issue No.	Description	Status: Resolved	Status: Open
1.	Use of acceleration drag volumes which differ from those approved in the AC to determine drag on sharp cornered structures.	X	
2.	Phasing of load harmonics used to analyze structures affected by CO and post-chug loads.	X	
3.	FSI methodology used for CO and chugging submerged structure loads.	X	
4.	Use of calibration factors developed from Monticello in-plant tests for use in defining SRV submerged structure drag loads.	X	

3.1 Acceleration Drag Volumes for Sharp Cornered Structures

The Acceptance Criteria 2.14.2 section 2b in NUREG-0661 states that drag forces on structures with sharp corners (e.g. rectangles and "I" beams) must be computed by considering forces on an equivalent cylinder of diameter $D_{eq}=2^{1/2}\ L_{max}$ where L_{max} is the maximum transverse dimension. The intent of this criterion is to provide a conservative bound (based on very limited data) that includes non-potential flow effects such as vortex shedding on both the acceleration drag due to hydrodynamic mass and the "standard" drag proportional to velocity squared. Since the dominant load for the Ring Beam (the primary non-cylindrical structure) is acceleration drag, the issue concerns only the hydrodynamic mass or acceleration volume and not the drag coefficient in the Hope Creek plant-specific case.

The PUAR states that "published" acceleration drag volumes listed in Table 1-4.1-1 are used for sharp edged structures rather than the equivalent cylinder specified in the acceptance criteria. The detailed response in Reference 7 to a Request for Information (Item 8) explains that modeling of the actual structures is necessary.

For the in-plane direction, the PUAR methodology models the ring girder as an I beam. While the acceleration volumes thus obtained are less than those given by the conservative AC methodology, the PUAR uses an interference coefficient of 2.0, which is very conservative in this direction . Thus, an adequate margin is provided for any possible non-potential flow effects. In the out-of-plane direction, the PUAR method calculates the acceleration volumes of the ring girder load on the hydrodynamic volume of a circumscribed rectangle combined with the actual volume of the ring girder. Again, the acceleration volumes obtained are less than those calculated by the conservative AC methodology. The use of an interference coefficient of 2.0 in the PUAR for this direction is also conservative, but not by as large a margin as for in-plane loads. However, the applicant has stated that the out-of-plane drag loads can be increased 31% without exceeding allowable stresses for the critical load combinations. Since the flow is expected to be very nearly potential in the parameter range of CO and post-chug acceleration spectrum where the major energy is concentrated, this should provide an adequate margin to cover any non-potential flow effects. An additional source of conservatism in the PUAR analysis is the use of a single mode dynamic load factor. We feel that these conservatisms in the interference corrections and the load application adequately compensate for any possible non-conservatism of the acceleration volumes.

On the above basis, BNL concludes that while the direct use of "published" acceleration volumes for sharp edge structures may not in general lead to conservative loads, the PUAR methodology for the application of these loads to the relevant structures, has sufficient conservatism to bound any hydrodynamically produced stresses that could arise in these structures.

3.2 CO and Post-Chug Harmonic Phasing

The DBA condensation oscillation and the post-chug load definitions on the torus shell and on submerged structures, accepted in the NUREG-0661, were based on data from a series of blowdowns in the FSTF facility (NEDE-24539), subject to additional confirmatory tests reported in the General Electric Letter Report M1-LR-81-01 of April 1981.

The condensation oscillation load definition as described in NEDO-21888 is based on taking the absolute sum of 1 Hertz components of a spectrum from 0 to 50 Hz. Three alternative spectra are to be calculated with the one producing maximum response used for load definition. The procedure was found acceptable in Supplement No. 1 to NUREG-0661, dated August 1982, because the demonstrated high degree of conservatism associated with the direct summation of the Fourier components of the spectrum was sufficient to compensate for any uncertainties concomitant with the data available. The post-chug load definition is based on bounding FSTF chugging data but otherwise follows similar procedures to those used in the CO load definition.

The PUAR uses a factor of .65 to multiply the CO and post-chug loads computed on the basis of the absolute sum of the harmonic components. The justification is based on comparisons of measured and predicted stresses in the FSTF facility using statistical studies of different phasing models (References 8, 9, 10, 11). The factor .65 is chosen to give 84% non-exceedance probability with a confidence level of 90%. The PUAR does use an additional spectrum, Alternate 4, for the CO loading, based on test M12 from the supplementary tests reported in the letter report M1-LR-81-01. The information in Table 1-4.1-4 of the PUAR provides additional justification to show that the computed loads (using the .65 factor and Alternates 1 through 3) bound the measured stresses at critical points in the FSTF facility by 11% for axial shell stress to 69% for column force. The use of Alternate 4 in the Hope Creek plant provides an additional conservatism of about 20% to the shell response.

The procedures are a conservative application of the phasing design rules evaluated in Reference 12 and are therefore found acceptable.

3.3 FSI Methodology for CO and Chugging Drag Loads

A detailed discussion of the method used to account for FSI effects on condensation oscillation and chugging submerged structure loads is provided in Reference 13. The methodology described in this note is used to compute acceleration fields across a submerged structure anywhere in the torus resulting from FSI, based on knowing the torus boundary acceleration. The method is presented as an alternative to the NRC Acceptance Criteria suggestion of adding the boundary accelerations directly to the local fluid acceleration to account for FSI effects since the latter is deemed too conservative.

The review of the method outlined in Reference 13 has shown it to be reasonable and acceptable. The equations derived for fluid accelerations and pressure fields are plausible approximations for the conditions prevailing in the suppression pool. Assumed boundary conditions including the driving one at the torus wall are suitable. Overall trends as well as the acceleration fields depicted in the selected results appear reasonable. Therefore, the alternate procedure used to account for FSI effects on submerged structures is considered acceptable in this application.

3.4 Calibration of SRV Drag Loads Based on In-Plant Tests

For other NUTECH plants BNL requested clarification of the detailed procedures used to derive the calibration factors from in-plant tests for SRV submerged-structure loads. On the basis of the response to those other PUAR reviews of NUTECH plants, BNL considers the procedures as an acceptable modification of the AC.

The SRV bubble pressure data from Monticello tests is shown to be bounded using a bounding factor of 1.75 instead of the 2.5 specified in the AC. In the Hope Creek plant, dynamic load factors are derived on the basis of Monticello in-plant tests.

BNL considers these procedures to be a reasonable application of the in-plant test results, and considers any potential uncertainties associated with the limited data base to be bounded by other conservatisms associated with the design load calculation procedures.

4. CONCLUSIONS

A post-implementation pool dynamic load audit of the Hope Creek PUAR has been completed to verify compliance with the generic acceptance criteria of NUREG-0661. Four major differences between the PUAR and the AC were identified along with some other minor issues needing additional clarification. Based on additional information supplied by the applicant, as detailed in the previous section, all of these issues were resolved. The review of the Hope Creek PUAR has been completed with no issues or concerns outstanding.

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APPENDIX O

ELIMINATION OF ARBITRARY INTERMEDIATE
PIPE BREAKS

TABLE OF CONTENTS

HOPE CREEK GENERATING STATION UNIT 1
SAFETY EVALUATION FOR THE ELIMINATION OF ARBITRARY
INTERMEDIATE PIPE BREAKS

I INTRODUCTION

In the "Background" to Branch Technical Position (BTP) MEB 3-1 as presented in Standard Review Plan (SRP) Section 3.6.2 (NUREG-0800), the staff position on pipe break postulation acknowledged that pipe rupture is a rare event that may only occur under unanticipated conditions such as those that might be caused by possible design, construction, or operation errors, unanticipated loads, or unanticipated corrosive environments. The BTP MEB 3-1 pipe break criteria were intended to utilize a technically practical approach to ensure that an adequate level of protection had been provided to satisfy the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. Specific guidelines were developed in BTP MEB 3-1 to define explicitly how the requirements of GDC 4 were to be implemented. The SRP guidelines in BTP MEB 3-1 were not intended to be absolute requirements but rather represent viable approaches considered to be acceptable by the staff.

The SRP provides a well-defined basis for performing safety reviews of light-water reactors. The uniform implementation of design guidelines in BTP MEB 3-1 ensures that a consistent level of safety will be maintained during the licensing process. Alternative criteria and deviations from the SRP are acceptable provided an equivalent level of safety can be demonstrated. Acceptable reasons for deviations from SRP guidelines include changes in emphasis of specific guidelines as a result of new developments from operating experience or plant-unique design features not considered when the SRP guidelines were developed.

The SRP presents the most definitive basis available for specifying NRC's design criteria and design guidelines for an acceptable level of safety for reviews of light-water-reactor facilities. The SRP guidelines resulted from many years of experience gained by the staff in establishing and using regulatory requirements in the safety evaluation of nuclear facilities. The SRP is part of a continuing regulatory standards development activity that not only documents current methods of review, but also provides a basis for an orderly modification of the review process when the need arises to clarify the content, correct any errors, or modify the guidelines as a result of technical advancements or an accumulation of operating experience. Proposals to modify the guidelines in the SRP are considered for their impact on matters of major safety significance.

The staff has recently received a request from the applicant for Hope Creek Generating Station, Unit 1, to consider an alternative approach to the guidelines in SRP Section 3.6.2, BTP MEB 3-1, regarding the postulation of intermediate pipe breaks (Mittl, June 11, 1985). For all high energy piping systems identified in that request, the applicant proposes to eliminate from design considerations those breaks generally referred to as "arbitrary intermediate breaks" (AIBs), which are defined as those break locations that, on the basis of piping stress analysis results, are below the stress and fatigue limits specified in BTP MEB 3-1, but are selected to provide a minimum of two postulated breaks between the terminal ends of a piping system. The applicant has stated that occupational radiation exposure during inspection, maintenance, and repair will be reduced over the life of the plant. The applicant is requesting approval of alternative pipe break criteria to provide the flexibility to remove or not to shim restraints in the future, if deemed necessary. However, the applicant has stated that the elimination of AIBs will not impact the environmental qualification of safety-related equipment. The break postulation for environmental effects is performed independently of break postulation for pipe whip and jet impingement.

In the early 1970s when the pipe break criteria in BTP MEB 3-1 were first drafted, the advantages of maintaining low stress and usage factor limits were clearly recognized, but it was also believed that equipment in close proximity to the piping throughout its run might not be adequately designed for the environmental consequences of a postulated pipe break if the break postulation proceeded on a purely mechanistic basis using only high stress and terminal end breaks. As the pipe break criteria were implemented by the industry, the impact of the pipe break criteria became apparent on plant reliability and costs as well as on plant safety. Although the overall criteria in BTP MEB 3-1 have resulted in a viable method that ensures that adequate protection has been provided to satisfy the requirements of GDC 4, it has become apparent that the particular criterion requiring the postulation of AIBs can be overly restrictive and may result in an excessive number of pipe rupture protection devices which do not provide a compensating level of safety.

At the time the BTP MEB 3-1 criteria were first drafted, high energy leakage cracks were not being postulated. In Revision 1 to the SRP (NUREG-0800), the concept of using high energy leakage cracks to mechanistically achieve the environment desired for equipment qualification was introduced to cover areas that are below the high stress/fatigue limit break criteria and that would otherwise not be enveloped by a postulated break in a high energy line. In the proposed elimination of AIBs, the staff believes that the essential design requirement of equipment qualification is not only being retained but is being improved, since all safety-related equipment is to be qualified environmentally. Furthermore, certain elements of construction that may lead to reduced reliability are being eliminated.

In addition, some requirements that have developed over the years as part of the licensing process have resulted in additional safety margins that overlap the safety margin provided in the pipe break criteria. For example, the criteria in BTP MEB 3-1 include margins to account for the possibility of flaws that might remain undetected in construction and to account for unanticipated piping steady-state vibratory loadings not readily determined in the design process. However, inservice inspection requirements for the life of the plant to detect flaws before they become critical and staff positions on the vibration monitoring of safety-related and high energy piping systems during preoperational testing further reduce the potential for pipe failures occurring from these causes.

Because of the recent interest expressed by the industry to eliminate the AIB criteria and, particularly, in response to the submittals provided by several utilities including PSE&G, the staff has reviewed the BTP MEB 3-1 pipe break criteria to determine where such changes may be made.

II BASES FOR THE ELIMINATION OF ARBITRARY INTERMEDIATE PIPE BREAKS

In the letter dated June 11, 1985, the applicant requested the elimination of AIBs and the technical bases for the proposal. The consensus in the nuclear industry is that current knowledge and experience support the conclusion that designing for the AIBs is not justified. The reasons for this conclusion are discussed in the following paragraphs.

(1) Operating Experience not Supporting Need for Criteria

The combined operating history of commercial nuclear plants (extensive operating experience in over 80 operating U.S. plants and a number of similar plants overseas) has not shown the need to provide protection from the dynamic effects of AIBs.

(2) Piping Stresses Well Below ASME Code Allowable Values

Currently, AIBs are postulated to provide a minimum of two pipe breaks at the two highest stress locations between piping terminal ends. Consequently, AIBs are postulated at locations in the piping system where pipe stresses and/or cumulative usage factors are well below ASME Code allowable values. Such postulation necessitates the installation and maintenance of complicated mitigating devices to afford protection from dynamic effects such as pipe whip and/or jet impingement. When these selected break locations have stress levels only slightly greater than the rest of the system, installation of mitigating devices lends little to enhance overall plant safety.

(3) Unanticipated Thermal Expansion Stress

Unanticipated stresses resulting from restraint of thermal expansion can be introduced into the piping system if pipe rupture protection devices come into contact with the pipes. The potential for this happening is greater than that for mechanistic failure at an arbitrary break point. To prevent a consequent decrease in the overall reliability of the pipe system, an additional as-built verification step is involved in the design process for each installed pipe whip restraint. Elimination of AIBs would significantly reduce the effort involved in designing and installing pipe rupture protection devices.

(4) Access

Access during plant operation for maintenance and inservice inspection activities can be improved because of the elimination of congestion created by these pipe rupture protection devices and the supporting structural steel associated with arbitrary pipe breaks.

(5) Reduction in Radiation Exposure

In addition to the decrease in maintenance effort, a corresponding reduction in person-rem exposure can be realized because fewer person-hours will be spent in radiation areas, per the as low as is reasonably achievable criterion.

(6) Decrease in Heat Loss

The elimination of pipe whip restraints associated with arbitrary breaks will preclude the requirement for cutback insulation or special insulating assemblies near the close-fitting restraints. This will reduce the heat loss to the surrounding environment, especially inside containment.

III STAFF EVALUATION OF THE BASES FOR THE ELIMINATION OF ARBITRARY BREAKS

The technical bases for the elimination of the AIB criteria as discussed in the preceding section of this report provided many arguments supporting the applicant's conclusion that the current SRP guidelines on this subject should be changed. However, it is not apparent that a unilateral position by the utility concluding an unconditional deletion of the AIB criteria can be justified without a clear understanding of the safety implications that may result for the various classes of high energy piping systems involved. In this section, the staff will discuss the bases for the current AIB criteria from an ASME Code design standpoint and put into perspective the uncertainty factors on which the need to postulate AIBs should be evaluated.

Although the ASME Code design requirements for Class 1 piping systems differ from those for Class 2 and 3 piping systems, there are other design considerations that are common to Class 1, 2, and 3 systems. These other design considerations (namely (1) intergranular stress corrosion cracking, (2) water/steam hammer, and (3) thermal fatigue) can affect the safety of the systems in which AIBs are eliminated. Therefore, while evaluating the acceptabiltity of the applicant's proposed deviation from SRP Section 3.6.2, the staff has examined the significance of the above three additional design considerations for the specific Hope Creek piping systems proposed by the applicant for elimination of AIBs.

ASME Code, Class 1 Piping Systems

In accordance with BTP MEB 3-1 (Paragraph B.1.c.(1)), breaks in ASME Code, Class 1 piping should be postulated at the following locations in each piping and branch run:

(a) at terminal ends;

(b) at intermediate locations where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or (13) of ASME Code NB-3650 exceeds 2.4 Sm;

(c) at intermediate locations where the cumulative usage factor exceeds 0.1.

(d) If two intermediate locations cannot be determined by (b) and (c) above, two highest stress locations based on Eq. (10) should be selected.

The AIB criteria are stated in (d) above. It should be noted that the request for alternative criteria does not propose to deviate from the criteria in (a), (b), and (c) above. Pipe breaks will continue to be postulated at terminal ends irrespective of the piping stresses.

Pipe breaks are to be postulated at intermediate locations where the maximum stress range as calculated by Equation (10) and either Equation (12) or (13) exceeds 2.4 Sm. The stress evaluation in Equation (10) represents a check of the primary plus secondary stress intensity range resulting from ranges of pressure, moments, thermal gradients, and combinations thereof. Equation (12) is intended to prevent formation of plastic hinges in the piping system caused only by moments resulting from thermal expansion and thermal anchor movements. Equation (13) represents a limitation for primary plus secondary membrane plus bending stress intensity excluding thermal bending and thermal expansion stresses; this limitation is intended to ensure that the K_e factor (strain concentration factor) is conservative. The K_e factor was developed to compensate for the absence of elastic shakedown when primary plus secondary stresses exceed 3 Sm.

With respect to piping stresses, the pipe break criteria were not intended to imply that breaks will occur when the piping stress exceeded 2.4 Sm (80% of the primary plus secondary stress limit). It is the staff's belief, however, that if a pipe break were to occur (on one of those rare occasions), it is more likely to occur at a piping location where there is the least margin to the ultimate tensile strength.

Similarly, from a fatigue strength standpoint, the staff believes that a pipe break is more likely to occur where the piping is expected to experience large cyclic loadings. Although the staff concurs with the industry belief that a cumulative usage factor of 0.1 is a relatively low limit, the uncertainties involved in the design considerations with respect to the actual cyclic loadings experienced by the piping tend to be greater than the uncertainties involved in the design considerations used for the evaluation of primary and secondary stresses in piping systems. The staff finds that the conservative fatigue considerations in the current SRP guidelines provide an appropriate margin of safety against uncertainties for those locations where fatigue failures are likely to occur (e.g., at local welded attachments).

ASME Code, Class 2 and 3 Piping Systems

In accordance with BTP MEB 3-1 (Paragraph B.1.c.(2)), breaks in ASME Code, Class 2 and 3 piping should be postulated at the following locations:

(a) at terminal ends

(b) at intermediate locations selected by one of the following criteria:

  (i) at each pipe fitting, welded attachment, and valve

  (ii) at each location where the stresses exceed 0.8 ($1.2\ S_h + S_A$) but at not less than two separated locations chosen on the basis of highest stress.

In the submittal, the applicant has not proposed changing Criterion (a) above. Postulation of pipe breaks at terminal ends will not be eliminated in the proposed SRP deviation for Class 2 and 3 piping systems.

The AIB criteria is stated in (b)(ii) above. Breaks are to be postulated at intermediate locations where the stresses exceed 0.8 ($1.2\ S_h + S_A$) but "at not less than two separated locations chosen on the basis of highest stress." The stress limit provided in the above pipe break criterion represents the stress associated with 80% of the combined primary and secondary stress limit. Thus, a break is required to be postulated where the maximum stress range as calculated by the sum of Equations (9) and (10) of Paragraph NC/ND-3652 of the ASME Code, Section III, exceeds 80% of the combined primary and secondary stress limit, when considering those loads and conditions for which level A and level B stress levels have been specified in the system's design specification (i.e., sustained loads, occasional loads, and thermal expansion) including an operating basis earthquake (OBE) event. However, the Class 2 and 3 pipe break criteria do not provide for the postulation of pipe breaks based on a fatigue limit because an explicit fatigue evaluation is not required in the ASME Code for these classes of construction because of favorable service experience and lower levels of operating cyclic stresses.

For those Class 2 and 3 piping systems that experience a large number of stress cycles (e.g., main steam and feedwater systems), the ASME Code has provisions that are intended to address these types of loads. The rules governing considerations for welded attachments in ASME Code, Class 2 and 3 piping which do preclude fatigue failure are partially given in Paragraph NC/ND-3645 of the ASME Code. The Code states:

> External and internal attachments to piping shall be designed so as not to cause flattening of the pipe, excessive localized bending stresses, or harmful thermal gradients in the pipe wall. It is important that such attachments be designed to minimize stress concentrations in applications where the number of stress cycles, due either to pressure or thermal effect, is relatively large for the expected life of the equipment.

Code rules governing the fatigue effects associated with general bending stresses caused by thermal expansion are addressed in Paragraph NC/ND-3611.2(e) and are generally incorporated into the piping stress analyses in the form of an allowable stress reduction factor.

Thus, it can be concluded that when the piping designers have appropriately considered the fatigue effects for Class 2 and 3 piping systems in accordance with Paragraph NC/ND-3645, the likelihood of a fatigue failure in Class 2 and 3 piping caused by unanticipated cyclic loadings can be significantly reduced.

Additional Design Considerations

In its presentation to the Advisory Committee on Reactor Safeguards on June 9, 1983, and in an October 5, 1983, meeting of a group of pressurized-water-reactor (PWR) near-term operating license utilities and the NRC staff, the staff indicated that the elimination of AIBs was not to apply to piping systems in which stress corrosion cracking, large unanticipated dynamic loads such as steam or water hammer, or thermal fatigue in fluid-mixing situations could be expected to occur. In addition, the elimination of AIBs was to have no effect on the requirement to environmentally qualify safety-related equipment. In fact, this requirement was to be clarified to ensure positive qualification requirements.

(1) Intergranular Stress Corrosion Cracking

At Hope Creek, the applicant has taken steps to minimize the potential for intergranular stress corrosion cracking (IGSCC) in high energy lines. The IGSCC potential is likely to be reduced if the following factors are controlled: high residual tensile stresses, susceptible piping material, and a corrosive environment. The NRC Piping Review Committee (NUREG-1061, Vol. 5, April 1985) has indicated the type of materials that are considered resistant to IGSCC. For example, stainless steel types 304L, 308L, and 316L are considered resistant to IGSCC. In addition, certain treatments given to the materials will make them resistant to IGSCC. Also, certain mitigating processes applied to the welds may reduce the likelihood of IGSCC.

The applicant has reported in the June 11, 1985, letter that only a low-carbon-content stainless steel (type 304L) has been used in the portion of the residual heat removal system connecting to the recirculation system. The remainder of the affected system piping is ferritic carbon steel, which has been found not to be susceptible to IGSCC. Furthermore, the applicant has taken steps to minimize the existence of a corrosive environment by specifying stringent criteria for internal and external cleaning and by controlling the water chemistry during power ascension and normal operation.

NUREG-1061 (Vol. 5) indicates that, if any unanticipated severe conditions should occur, the break would most likely be located at terminal ends, at connections to components, and at other locations that introduce higher stress concentration or that exceed the stated threshold limits specified in SRP Section 3.6.2. Because breaks are postulated for these locations, the staff concurs with the applicant's conclusion that elimination of AIBs would not introduce adverse effects.

(2) Water/Steam Hammer

According to NUREG-0927, boiling-water-reactor (BWR) plants report a higher frequency of water/steam hammer events than PWR plants primarily because of two factors: line voiding and presence of steam-water interfaces in BWRs. Line voiding was the largest single cause of BWR water hammers and was responsible for at least 39 of the 69 unanticipated water hammer events in BWR plants that were reported from 1969 through mid-1981. NUREG-0927 also reports that the addition of keep-full systems to BWR plants has reduced the frequency of water hammers. Keep-full systems continuously supply water to idle lines to prevent voiding.

The applicant has incorporated several water-hammer-minimization features into piping design operations at Hope Creek. The discharge lines of the residual heat removal system, low pressure coolant injection system, high pressure coolant injection (HPCI) system, core spray system, and reactor core isolation cooling (RCIC) system are maintained in a full condition. They are kept full up to the injection isolation valves by jockey pumps. Beyond the injection isolation valves, the line is not drained when the system is on standby, thus, maintaining the discharge lines full (Mittl, August 9, 1985). The feedwater system is started with flow initially through bypass and recirculation lines to avoid water hammer during startup. During operation the lines will remain filled thus minimizing the potential for water hammer. The reactor water cleanup system is continuously in operation to purify the reactor water, and the lines will be kept full thus minimizing the potential for water hammer.

The applicant has reported that the main steam, HPCI, and RCIC steam lines that experience transients as a result of fast valve closure have been designed to accommodate the effects of these loadings (Mittl, June 11, 1985). The steam supply lines are sloped to allow moisture collecting in the lines to drain to a collecting pot. The main steam isolation valve drain lines are sloped so that any condensate collecting in the lines will drain to the condenser.

As stated in the letter dated August 9, 1985, the applicant has committed to conduct piping preoperational and startup testing for steam and water hammer. The staff concurs with the applicant's conclusion that the design features and operating procedures described above will minimize the potential for water/steam hammer occurrence in several systems discussed above.

(3) Thermal Fatigue

The applicant has concluded, and the NRC staff concurs, that the systems for which AIBs are to be eliminated are not susceptible to thermal fatigue and mixing for the following reasons:

(a) The fatigue analysis performed by the applicant for all Class 1 piping systems shows that all of the Class 1 AIB locations involve cumulative usage factors well below the AIB postulation limit of 0.1 (Mittl, July 3, 1985). For Class 2 and 3 piping components, fatigue failure protection is ensured by the allowable stress range checks and a stress range reduction factor for thermal expansion stress. The mandatory breaks are postulated at 80% of the Code allowable stresses, even after the AIBs identified in the June 11, 1985, submittal have been eliminated.

(b) The applicant has minimized the cyclic thermal stresses and the resultant thermal fatigue in the Hope Creek piping systems by limiting the mixing of low-velocity, low-temperature water with high-temperature water. The piping systems for which AIBs are to be eliminated will not exhibit temperature gradients caused by flow stratification (Mittl, July 3, 1985). The applicant has come to this conclusion on the basis of a review that showed that one or more of the following conditions exist:

- The affected pipes have no flow during normal plant operation (e.g., HPCI and RCIC).

- The piping layout consists of vertical runs or sloped horizontal runs with valves and fittings to promote mixing.

- The piping is preheated (e.g., HPCI and RCIC steam supply lines) to minimize thermal stresses during system initiation.

Evaluation of Class 1 Piping Systems

For Class 1 piping, a considerable amount of quality assurance in design, analyses, fabrication, installation, examination, testing, and documentation is provided which ensures that the safety concerns associated with the uncertainties discussed above are significantly reduced. On the basis of the staff evaluation of the design considerations given to Class 1 piping, the stress and fatigue limits provided in the BTP MEB 3-1 break criteria, and the relatively small degree of uncertainty in unanticipated loadings, the staff finds that the need to postulate AIBs in ASME Code, Class 1 piping in which large unanticipated dynamic loads, stress corrosion cracking, and thermal fatigue such as in mixing situations are not present and in which all equipment has been environmentally qualified is not compensated for by an increased level of safety. In addition, systems may actually perform more reliably for the life of the plant if the SRP criterion to postulate AIBs for ASME Code, Class 1 piping is eliminated. The staff has concluded that the above described requirements are present for those ASME Code, Class 1 piping systems identified in the applicant's submittal of June 11, 1985.

Evaluation of Class 2 and 3 Piping Systems

On the basis of the staff evaluation of the design considerations given to Class 2 and 3 piping, the stress limits provided in the SRP break criterion, and the relatively small degree of uncertainty in unanticipated loadings, the staff finds that dispensing with AIBs is justified for Class 2 and 3 piping in which stress corrosion cracking, large unanticipated dynamic loads, or thermal fatigue in fluid mixing situations are not expected to occur provided (1) the piping designers have appropriately considered the effects of local welded attachments per Paragraph NC/ND-3645, and (2) all safety-related equipment in the vicinity of Class 2 and 3 piping systems has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break with the greatest consequences on the equipment. The staff has concluded that the above described requirements are present for those ASME Code, Class 2 and 3 piping systems identified in the applicant's letter dated June 11, 1985.

Piping Systems not Included in Proposal

For those piping systems, or portions thereof, that are not included in the applicant's submittal of June 11, 1985, the staff requires that the existing guidelines in BTP MEB 3-1 of the SRP (NUREG-0800) Revision 1, be met. However, should other piping lines that are not specifically identified in the applicant's submittal subsequently qualify for the conditions described above, the implementation of the proposed elimination of the AIB criteria may be used, provided those additional piping lines are appropriately identified to the staff.

Conclusion

The applicant has proposed a deviation from the current SRP guidelines by requesting relief from postulating AIBs in high energy piping systems that are not susceptible to intergranular stress corrosion cracking, steam or water hammer effects, and thermal fatigue in fluid mixing. The SRP guideline that requires that two intermediate breaks be postulated even when the piping stress is low resulted from the need to ensure that equipment qualified for the environmental consequences of a postulated pipe break was provided over a greater portion of the high energy piping run. This proposal is based, in part, on the condition that all equipment in the spaces traversed by the fluid system lines, for which AIBs are being eliminated, is qualified for the environmental (nondynamic) conditions that would result from a nonmechanistic break with the greatest consequences on surrounding equipment. In addition, the applicant has committed to perform preoperational testing of all the systems identified in the June 11, 1985, submittal and also to monitor those systems for vibration during preoperational and startup testing.

The staff has evaluated the technical bases for the proposed deviation with respect to satisfying the requirements of GDC 4. Furthermore, the staff has considered the potential problems identified in NUREG/CR-2136 that could impact overall plant reliability when excessive pipe whip restraints are installed. On the basis of its review, the staff finds that when those piping system conditions as stated above are met, there is a sufficient basis for concluding that an adequate level of safety exists to accept the proposed deviation.

Thus, on the basis of the piping systems having satisfied the above conditions, the staff concludes that the pipe rupture postulation and the associated effects are adequately considered in the design of Hope Creek Generating Station, Unit 1, and, therefore, the deviation from the SRP is acceptable.

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10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
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11a. TYPE OF REPORT
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b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES
Pertains to Docket No. 50-354

13. ABSTRACT (200 words or less)

Supplement No. 3 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 and Supplements 1 and 2.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

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