NUREG-1048
Supplement No. 2

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

August 1985

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ABSTRACT

Supplement No. 2 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In October 1984, the U.S. Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-1048) on the application filed by Public Service Electric and Gas Company (PSE&G) (applicant), acting on behalf of itself and Atlantic City Electric Company, for a license to operate the Hope Creek Generating Station (Docket No. 50-354). At that time, the staff identified items that were not yet resolved with the applicant. Supplement No. 1 to the SER was was issued in March 1985. 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 K presents the Safety Evaluation Report pertaining to TMI Action Plan Item II.D.1, "Safety and Relief Valve Testing." Appendix L presents the Safety Evaluation Report pertaining to TMI Action Plan Item II.E.4.2, "Demonstration of Containment Purge and Vent Valve Operability." Appendix M presents a report prepared for NRC by EG&G, Idaho, Inc., "Conformance to Regulatory Guide 1.97, Hope Creek Generating Station."

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.

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

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1.7 Outstanding Issues

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

1.8 Confirmatory Issues

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

Table 1.2 Outstanding issues (revised from SSER 1)

Issue	Status	SER section(s)
(1) Riverborne missiles	Under review	
(2) Equipment qualification	Partial closure	3.10
(3) Preservice inspection program	Awaiting information	
(4) GDC 51 compliance	Closed	6.2.7
(5) Solid-state logic modules	Awaiting information	
(6) Postaccident monitoring instrumentation	Closed	7.5.2.3
(7) Minimum separation between non-Class 1E conduit and Class 1E cable trays	Under review	
(8) Control of heavy loads	Closed	9.1.5 (Supplement 1)
(9) Alternate and safe shutdown	Partial closure	9.5.1.4
(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
(b) Requalification training programs	Closed	13.2.1.2
(c) Replacement training programs	Closed	13.2.1.3
(d) TMI issues I.A.2.1, I.A.3.1, and II.B.4	Closed	13.2.1.4
(e) Nonlicensed training programs	Closed	13.2.2
(13) Emergency dose assessment computer model	Under review	
(14) Procedures generation package	Awaiting information	
(15) Human factors engineering	Under review	

Table 1.3 Confirmatory issues (revised from SSER 1)

Issue	Status	SER section(s)
(1) Feedwater isolation check valve analysis	Awaiting information	
(2) Plant-unique analysis report	Under review	
(3) Inservice testing of pumps and valves	Awaiting information	
(4) Fuel assembly accelerations	Closed	4.2
(5) Fuel assembly liftoff	Closed	4.2
(6) Review of stress report	Awaiting information	
(7) Use of Code cases	Closed	5.2.1.2
(8) Reactor vessel studs and fasteners	Under review	
(9) Containment depressurization analysis	Under review	
(10) Reactor pressure vessel shield annulus analysis	Under review	
(11) Drywell head region pressure response analysis	Under review	
(12) Drywell-to-wetwell vacuum breaker loads	Under review	
(13) Short-term feedwater system analysis	Under review	
(14) Loss-of-coolant-accident analysis	Closed	6.3.5, 15.9.3
(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	
(19) Reactor mode switch	Under review	
(20) Engineered safety features reset controls	Awaiting information	

Table 1.3 (Continued)

Issue	Status	SER section(s)
(21) High pressure coolant injection initiation	Awaiting information	
(22) IE Bulletin 79-27	Under review	
(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	Under review	
(26) Multiple control system failures	Under review	
(27) Relief function of safety/relief valves	Under review	
(28) Main steam tunnel flooding analysis	Under review	
(29) Cable tray separation testing	Under review	
(30) Use of inverter as isolation device	Under review	
(31) Core damage estimate procedure	Awaiting information	
(32) Continuous airborne particulate monitors	Awaiting information	
(33) Qualifications of senior radiation protection engineer	Closed	12.5.1
(34) Onsite instrument information	Awaiting information	
(35) Airborne iodine concentration instruments	Under review	
(36) Emergency Plan items	Under review	
(37) TMI Item II.K.3.18	Closed	15.9.3

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

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

3.10.2 Pump and Valve Operability Assurance

In Section 3.10.2 of the SER, TMI Item II.D.1, "Performance Testing of Boiling Water Reactor and Pressurized Water Reactor Relief and Safety Valves," was identified as an issue that still had to be reviewed. The staff has completed its review of this issue, and the results are presented in Appendix K of this supplement. The staff finds that the applicant has acceptably demonstrated the ability of the reactor coolant system relief and safety valves to function under expected operating conditions for design-basis transients and accidents as defined in TMI Item II.D.1.

The staff has also completed its review of TMI Item II.E.4.2, "Containment Isolation Dependability." The results of this review are presented in Appendix L of this supplement. The staff finds that the dependability of containment isolation is demonstrated against the buildup of containment pressure resulting from a loss-of-coolant accident or a design-basis accident, with the restrictions that (1) during operating conditions 1, 2, and 3, all purge and vent valves will be sealed closed and under administrative control and (2) during power ascension and descension conditions, the 26-in. inboard valve will be used in series with the 2-in. bypass valve to control the release of containment pressure. Appropriate Technical Specifications should be issued that reflect which valves are to be sealed closed and under administrative control during the various operating modes. All valves should be verified to be closed at least every 31 days when in operating modes 1, 2, and 3.

4 REACTOR

4.2 Fuel System Design

Fuel Rod Mechanical Fracturing

In Section 4.2 of the SER, the staff stated that fuel rod mechanical fracturing is a confirmatory issue because the applicant had not provided the design limits. In its review, the staff assumed these limits to be similar to those for other boiling-water reactors (BWRs).

The applicant has submitted for staff review a plant-specific analysis using the approved methodology described in General Electric (GE) Topical Report NEDE-21175-3. The staff finds these results to be acceptable, and the issue of fuel rod mechanical fracturing is resolved.

Fuel Assembly Structural Damage From External Forces

In Section 4.2 of the SER, the staff identified fuel assembly structural damage from external forces as a confirmatory issue because the applicant had not provided the design limits. In its review, the staff assumed these limits to be similar to those for other BWRs.

The staff has approved GE Topical Report NEDE-21175-3 (letter from C. O. Thomas (NRC) to J. F. Quirk (GE), October 20, 1983), which describes an analytical method for evaluating seismic and loss-of-coolant-accident (LOCA) loads. The staff has also reviewed the Hope Creek plant-specific values of liftoff and acceleration. The results show that the vertical liftoff is less than the allowable liftoff limit given in NEDE-21175-3, which is referenced by the applicant, and the accelerations are within the evaluation-basis limits, thereby ensuring structural integrity and control rod insertibility during seismic and LOCA events. The staff, therefore, concludes that the issue of seismic and LOCA loads is satisfactorily resolved for Hope Creek.

5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With ASME Code and Code Cases

5.2.1.2 Applicable Code Cases

As noted in Section 5.2.1.2 of the SER, staff acceptance of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) cases in the construction of Hope Creek is contingent on the applicant complying with all those conditions that are imposed in addition to the conditions specified in each Code case identified in Regulatory Guides (RGs) 1.84 and 1.85 as conditionally acceptable. Those Code cases are used in the construction of ASME Code, Section III, Class 1 components within the reactor coolant pressure boundary (RCPB). Information detailing the application of these Code cases has been supplied in Amendments 7 (August 1984) and 9 (January 1985) to the Final Safety Analysis Report (FSAR) in response to staff Requests for Additional Information 210.9, 210.10, and 210.12. The staff has reviewed the responses to these requests and finds they are acceptable.

The staff concludes that compliance with the requirements of these Code cases and the additional conditions imposed by RGs 1.84 and 1.85 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 General Design Criterion (GDC) 1, and is, therefore, acceptable.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.3 Secondary Containment

In the SER, the staff noted that the applicant had performed an analysis of the post-LOCA pressure transient in the reactor enclosure and had determined the length of time for the pressure to reduce from atmospheric to -0.25 in. water gauge (WG). The assumptions used in this analysis are documented in the SER. On the basis of this analysis, it would take 168 sec with an inleakage of 278 standard cubic feet per minute (10% inleakage of secondary containment free volume) following a LOCA to draw down the secondary containment pressure to -0.25 WG. The applicant committed to verify the design inleakage rate and drawdown time by preoperational and periodic tests.

By letter dated May 1, 1985 (R. Mittl, PSE&G, to A. Schwencer, NRC), the applicant requested the staff's approval to use an inleakage rate of 100% per day and the associated drawdown time of 375 sec. The staff has reviewed the applicant's request and concurs that using an inleakage rate of 100% per day with an associated drawdown time of 375 sec is acceptable for use at Hope Creek. Section 15.6.5 of this supplement presents the radiological consequences of the increased secondary containment drawdown time.

The applicant will verify the design inleakage rate and drawdown time by preoperational and periodic tests. The staff will include a requirement for these periodic tests in the Technical Specifications. The staff finds this analysis and test commitment to be in conformance with Branch Technical Position CSB 6-3 and, therefore, acceptable.

6.2.7 Fracture Prevention of Containment Pressure Boundary

In Section 6.2.7 of the SER, the staff indicated that all ferritic components in the Hope Creek containment pressure boundary, except for 24-in. check valves 1F074 A and B, have sufficient fracture toughness to meet GDC 51. To provide reasonable assurance that these valves will comply with GDC 51, the staff indicated that the applicant must commit to an augmented inservice inspection (ISI) program. The SER indicated that the outer and inner valve body surface must be inspected at the first refueling outage and at other times when the valve is disassembled for maintenance. In a letter dated March 12, 1985 (R. L. Mittl, PSE&G, to A. Schwencer, NRC), the applicant indicated that the inspection program will be implemented at the first refueling outage and will include inspection of the entire surface of the valve bodies, both internal and external, by surface examination or other examinations acceptable to the staff. A more detailed program will be provided when the applicant submits the ISI program for Hope Creek, which is scheduled for submittal for staff review 6 months after the start of commercial operation. On the basis of these commitments, the staff concludes that there is reasonable assurance these valves will comply with the requirements of GDC 51.

6.3 Emergency Core Cooling System

6.3.5 Performance Evaluation

In the SER, the staff reported the results of a LOCA analysis for a lead plant that was stated by the applicant to be representative of Hope Creek. The SER also noted that the applicant had committed to supply a plant-specific LOCA analysis for Hope Creek before fuel loading.

The applicant provided a plant-specific LOCA analysis in FSAR Amendment 10 dated May 1985. This analysis included a spectrum of large and small pipe breaks and indicated that the most limiting break is a design-basis break in a recirculation suction pipe. As for the lead plant, an assumed failure of the Channel A dc source coincident with the break resulted in the worst single-failure condition. The plant-specific results demonstrate compliance with the requirements of 10 CFR 50.46 as shown in revised Table 6.2.

From its review, the staff concludes that the plant-specific LOCA analyses for Hope Creek are acceptable.

Table 6.2 Applicant's compliance with acceptance criteria for emergency core cooling system (revised from SER)

Criterion	Maximum value from break analyses	Allowable
Peak cladding temperature (°F)	2046	2200
Maximum cladding oxidation (%)	1.5	17
Maximum total hydrogen generation (%)	0.10	1

7 INSTRUMENTATION AND CONTROLS

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings

7.5.2.3 Postaccident Monitoring Instrumentation

The applicant was requested by Generic Letter 82-33 to provide a report to the NRC describing how the postaccident monitoring instrumentation follows the guidance of RG 1.97 as applied to emergency response facilities. The applicant responded to the generic letter by letter dated April 15, 1983 (R. Mittl, PSE&G, to A. Schwencer, NRC). The letter referred to FSAR Sections 1.8 and 7.5 and Table 7.5-1 for a review of the affected instrumentation. Postaccident monitoring instrumentation was identified as an open item in the SER. Subsequently, the staff requested additional information which was provided by the applicant's letter dated May 14, 1985 (R. Mittl, PSE&G, to W. Butler, NRC).

A detailed review and technical evaluation of the applicant's submittals were performed by EG&G Idaho, Inc., under contract to the NRC, with NRC supervision. This work resulted in Appendix M to this supplement, "Conformance to Regulatory Guide 1.97, Hope Creek Generating Station." This evaluation by EG&G Idaho concludes that the applicant either conforms to, or is justified in deviating from, the guidance of RG 1.97 for each postaccident monitoring variable. This issue is considered resolved.

9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

9.5.1.4 General Plant Guidelines

Safe Shutdown Capability

As part of the FSAR, the applicant provided Appendix 9A concerning fire protection for the safe shutdown capability. Further discussion of the safe shutdown capability including information on cable separation and the location of safe shutdown equipment is contained in FSAR Section 9.5.

The applicant's safe shutdown analysis states that systems needed for hot shutdown and cold shutdown are redundant and that one of the redundant systems needed for safe shutdown would be free of fire damage because it is provided separation, fire barriers, and/or alternative shutdown capability. To achieve hot shutdown, the main steam isolation and safety/relief valves and one of the following would be available: (1) the reactor core isolation cooling (RCIC) system, (2) the low pressure coolant injection (LPCI) system (either C or D), or (3) the core spray system, in addition to residual heat removal (RHR) system loop "A" or "B" in the suppression pool cooling mode, station auxiliary cooling system (SACS) loop "A" or "B," and station service water system (SSWS) loop "A" or "B." To go from hot shutdown to cold shutdown would require the "A" loops of the RHR system, the SACS, and the SSWS or the "B" loops of the RHR system, the SACS, and the SSWS. The safe shutdown review considered components, cabling, and support equipment for the systems identified above which are needed to achieve shutdown. The applicant has provided a cable separation review for all rooms of the plant housing safe shutdown equipment to ensure that at least one train of this equipment is available in the event of a fire in any of these rooms. The review identified the safety-related equipment and redundant safe shutdown system cabling and discussed the consequences of a fire in each of these rooms. The staff has reviewed the applicant's evaluation of the plant and concludes that it is an acceptable means of demonstrating that separation exists between safe shutdown system trains.

The applicant's review divided the areas according to the electrical division and assumed that a fire in one fire area also eliminated all equipment serviced by that division. Cables and equipment that were not required to go to cold shutdown were assumed to fail. No repairs were assumed. All cables and equipment of an electrical division other than the designated fire area division that are required to be operable were relocated to another fire area of a similar division or provided with other means of protection. The applicant has also indicated that alternative shutdown is required for the control room. If a fire should disable the control room, a remote shutdown panel located in a separate fire-protected room in the auxiliary building is provided as as alternative to

providing fire protection for the control room. Only Division II is electrically isolated from the control room. See the following section for a further discussion on the alternative shutdown capability.

The applicant has requested approximately 40 deviations from the fire protection requirements for safe shutdown capability. Because of the large number of deviations, the staff's review is ongoing and will be addressed in a future supplement to the SER.

Alternative and Dedicated Shutdown Capability

FSAR Section 7.4.2.4 describes the remote shutdown panel's design and capability. The present design objective of the remote shutdown panel is to achieve and maintain cold shutdown if an evacuation caused by a fire should disable the control room. The RCIC system, safety/relief valves, and one division of the RHR system, SACS, and SSWS can be controlled from the remote shutdown panel to achieve cold shutdown should a fire disable the control room. To ensure the availability of this remote shutdown panel in the event of a control room fire, switches are provided to transfer one division of shutdown capability to the remote shutdown panel and thereby provide electrical isolation between the control room and the remote shutdown panel.

The design of the remote shutdown panel complies with the performance goals outlined in Section III.L of Appendix R to 10 CFR 50. Reactivity control will be accomplished by a manual scram before the operator leaves the control room. The RCIC system will provide reactor coolant makeup, and the RHR system and the safety/relief valves will be used for reactor decay heat removal. Reactor vessel water level, reactor vessel pressure, suppression pool water level and temperature, RCIC pump turbine speed, and RHR system flow are among the instrumentation available at the remote shutdown panel. This panel will also include instrumentation and control of support functions needed for the shutdown equipment.

On the basis of the above, the staff concludes that the remote shutdown panel complies with the guidelines of Section III.L of Appendix R as contained in Paragraph C.5.c of Branch Technical Position CMEB 9.5-1 (NUREG-0800) and is, therefore, acceptable.

12 RADIATION PROTECTION

12.5 <u>Operational Radiation Protection Program</u>

12.5.1 Organization

Since the SER was issued, the applicant has reorganized the Radiation Protection Department at the Hope Creek Generating Station by creating two positions for Senior Radiation Protection Supervisors and one position for a Senior Radiological Engineer, all reporting directly to the Radiation Protection Manager (RPM). In the previous organizational structure, the persons filling these three basic positions reported to a Senior Radiation Protection Supervisor, who reported to the RPM. This new organizational structure allows more direct interaction between the RPM and the persons holding the three senior supervisory positions under him. In addition to ensuring the efficient operation of their respective groups, it is the responsibility of these three senior supervisory staff members to coordinate the work of radiation protection with the other organizations at the station. The staff finds this new organizational structure acceptable.

The applicant has stated that at least one member of the senior radiation protection supervisory staff shall meet or exceed the requirements of RG 1.8, "Personnel Selection and Training" (May 1977) for RPM. The individual assigned as the Senior Radiation Protection Supervisor/Operations should normally fulfill this requirement. This individual will also act as backup for the RPM during the manager's absence. In Section 12.5.1 of the SER, the staff stated the qualifications of the Senior Radiation Protection Supervisor would be compared to the qualifications of a backup RPM when they were submitted. The qualifications of the Senior Radiation Protection Supervisor/Operations at Hope Creek meet the requirements of Draft ANSI 3.1 of the American National Standards Institute for an individual temporarily filling the RPM position. On the basis of the applicant's submittal of the backup RPM's qualifications, this confirmatory item is resolved.

13 CONDUCT OF OPERATION

13.2 Training

13.2.1 Licensed Operator Training Program

13.2.1.1 Initial Training Program

In Section 13.2.1.1 of the SER, the staff had not completed the review of training for licensed personnel in the following areas: (1) structured observation training at an operating BWR; (2) cold license in-plant training, which includes in-plant training guidelines for licensed personnel; (3) prelicensing examination and testing, which includes Hope Creek classroom and simulator evaluations before licensing; (4) training on mitigating core damage for licensed personnel.

The applicant, in Amendment 9 to FSAR Appendix 13 K, has described the participatory training at an operating BWR for licensed operator and senior operator candidates. In addition to 2 weeks of observation at the Salem Generating Station, operator candidates will participate for 2 weeks at a comparable BWR plant. Senior operator candidates who have not previously held a BWR license will participate for 6 weeks at a comparable BWR, and those candidates who are scheduled to be senior shift supervisors will participate in a 6-month program at a comparable BWR. The observation/experience training will meet the guidelines of American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.1, 1981, Section 4.3.1; Generic Letter 84-10, "Administration of Operating Tests Prior to Initial Criticality," April 26, 1984; and Generic Letter 84-16, "Adequacy of On-Shift Operating Experience for Near Term Operating License Applicants," June 27, 1984.

The applicant has identified cold license operator in-plant training in Amendment 8 to FSAR Appendix 13 I. The program is designed to ensure that each candidate will receive practical work experience necessary to gain a thorough knowledge of the plant. The training will be documented in the form of individual in-plant training guidelines for reactor operator, senior reactor operator, and shift technical advisor candidates and is scheduled to include hot functional testing. The staff has reviewed the topics contained in the guidelines and concludes that they are appropriate for the intent of this segment of the training program.

With regard to prelicensing examination and testing, in Amendment 9 to FSAR Appendix J, the applicant has provided additional details on classroom presentations which include reviews of reactor theory, heat transfer, fluid mechanics, thermodynamics, health physics, Technical Specifications, and administrative procedures. Appendix J also states that simulator training will last 7 weeks and that classroom and simulator training will be equally divided. Because the applicant has a site-specific simulator, the staff concludes that licensed candidates will be adequately prepared for license examinations and will be able to maintain proficiency during the period between fuel loading and low-power testing or if unforeseen delays should occur.

In FSAR Amendment 9, the applicant has provided details of the training on mitigating core damage for licensed candidates. This training will include classroom and simulator exercises using normal, abnormal, and emergency operating procedures. The staff's review of the outline concludes that the training has addressed those topics in Enclosure 3 of the H. R. Denton letter of March 28, 1980, contained in NUREG-0737. The staff concludes that the applicant will provide an adequate training program and will meet the conditions in Section 13.2.1 of the Standard Review Plan, NUREG-0800.

13.2.1.2 Requalification Training Program

In Section 13.2.1.2 of the SER, the requalification training program was identified as an open item. The applicant, in a December 28, 1984, letter, submitted the Hope Creek NRC Licensed Operator Requalification Program. The program contains preplanned lecture series; on-the-job training that includes simulator exercises; annual written and oral examinations; special retraining programs; program evaluation; shift cycle training; instructor requalification; and program records. With the exception of shift cycle training, the applicant has stated that the program will be in effect within 3 months after an operating license is issued.

In an April 26, 1985, letter, the applicant submitted a revised requalification program, which

(1) stated that licensed operators in a senior operator upgrade program will receive presentations in related industry events, design changes, and areas involving plant operations

(2) identified the specific references used to structure lectures in the review of fundamentals and topics that may be included in operational proficiency training

(3) provided a revision of the section on shift cycle training to reflect completion of this training segment

(4) identified the standard for evaluating annual and biannual plant evolutions

(5) provided the method used to allow licensed personnel to be excused from attending a segment lecture series

The staff has reviewed the revised program and concludes that it meets the requirements of Appendix A of 10 CFR 55 and the conditions in H. R. Denton's letter of March 28, 1980, contained in NUREG-0737.

13.2.1.3 Replacement Training

In Section 13.2.1.3 of the SER, replacement training was identified as an open item. The applicant, in FSAR Amendment 9, has provided additional information on the training of replacement personnel. This training includes courses in nuclear plant fundamentals, plant systems training (including classroom and plant instruction), and operator practice (simulator and plant control room) and will take approximately 32 weeks. In addition, replacement senior operators will receive instruction in subjects related to their duties, which will include

reactor theory; handling and disposal of radioactive material; specific operating characteristics; core parameters; chemistry and radiochemistry; operating philosophy and use of administrative procedures; heat transfer, thermodynamics, and fluid flow; and responsibilities during emergency conditions. In response to Request for Additional Information (RAI) 630.4, the applicant has also stated that the necessary elements in the training on mitigating core damage will be included as a continuing process and will be contained in classroom presentations, simulator exercises, and in-plant demonstrations.

The staff has reviewed the outlines and concludes that the applicant will provide adequate training for replacement personnel.

13.2.1.4 TMI-Related Requirements for New Operating License

I.A.2.1 Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications

In the SER, the staff reviewed the initial training in heat transfer, fluid flow, and thermodynamics (HTFFT) contained in FSAR Appendices 13 A, C, and F. In Amendment 9 to FSAR Appendix 13 J, the applicant has provided additional details of the onsite review of HTFFT and included reviews of reactor theory, health physics, Technical Specifications, administrative procedures, and industry events related to operation. In addition, the applicant will provide 7 weeks of training on the Hope Creek simulator. The applicant has also provided training in the use of plant systems to control or mitigate an accident in which the core is severely damaged. The staff concludes that the applicant has provided the required training to meet the conditions of Item I.A.2.1 of NUREG-0737.

I.A.3.1 Revised Scope and Criteria for Licensing Exams

As specified in Item I.A.3.1 of NUREG-0660 and clarified in NUREG-0737, the staff requires that licensees prepare applicants for new examinations and develop and implement new examination criteria and lecture schedules for requalification programs. Specific requirements for new examinations include:

(1) All reactor operator license applicants shall take a written examination that includes a new category dealing with the principles of heat transfer and fluid mechanics. A time limit of 9 hours will be imposed for the written examination, and the passing grade is 80% overall and 70% in each category.

(2) All senior reactor operator license applicants shall take the reactor operator examination, an operating test, and a senior reactor operator written examination that includes a new category dealing with the theory of fluids and thermodynamics. A time limit of 7 hours will be imposed for the written examination, and the passing grade is 80% overall and 70% in each category.

(3) Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

(4) Simulator examinations will be included as part of the license examination.

Specific requirements related to requalification programs include:

(1) Contents of the licensed operator requalification program shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

(2) The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

(3) Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operations shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

The NRC requirements contained in NUREG-0660 and NUREG-0737 have been incorporated in NUREG-1021, "Operator Licensing Examiner's Standards," which was issued in October 1983 and revised in February 1985. Sections ES-202 and ES-403 of NUREG-1021 include a category on heat transfer, thermodynamics, and fluid flow; specify a revised time limit of 6 hours for the operator and senior operator written examinations; and state that the passing grade will remain 80% overall and at least 70% in each category.

Applicants for licenses are required to complete NRC Form 398, "Personal Qualifications Statement--Licensee," which provides permission to the NRC to inform facility management regarding the results of examinations. Because Hope Creek has a site-specific simulator in service, all licensed applicants will be administered a simulator examination as part of the license examination.

With regard to requalification programs, the applicant, in an April 26, 1985, letter, stated that the requalification program will provide fundamentals review lectures which include heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a damaged core. Participation in an accelerated training program is required if the written annual examination grade is less than 80% overall or less than 70% in a category. Accelerated training is also required if licensed personnel fail the annual in-plant walk-through or an operational test on the Hope Creek simulator. The requalification program also includes all the specific reactivity control manipulations that are contained in the March 28, 1980, letter from H. R. Denton to all licensees.

The staff has evaluated the applicant's response and concludes that it meets the intent of Item I.A.3.1 of NUREG-0737.

II.B.4 Training for Mitigating Core Damage

In the SER, training on mitigating core damage was an open item. In the response to RAI 630.4 in FSAR Amendment 9, the applicant has included training on mitigating core damage for shift technical advisors and operations personnel from the plant manager through the operations chain, including licensed operators. In addition, the applicant will provide training for managers and technicians in the Instrumentation and Control, Health Physics, and Chemistry Departments

on the use of installed instruments and systems to control and mitigate accidents that is commensurate with their duties and responsibilities. Emergency operating procedures are being developed and will be implemented in accordance with the procedures generation package, which has been submitted to the Commission. The staff has reviewed the outlines contained in response to RAI 630.4 and finds it meets the training provisions contained in Item II.B.4 of NUREG-0737.

13.2.2 Training for Nonlicensed Plant Staff

In the SER, the staff had not identified specific technical supervisory skills programs (TSSPs) for each management position in FSAR Section 13.1.2. In FSAR Amendment 9, the applicant has stated that nonlicensed personnel promoted to supervisory levels will complete a TSSP commensurate with their responsibilities. The staff is now aware that TSSPs are developed from elements contained in Section 5.2.1.8, "Training of Licensed Supervisors," of ANSI/ANS 3.1, 1981. The staff considers these elements appropriate for supervisors and concludes this item is resolved.

With regard to training of shift technical advisors, the staff had not completed the review of site-specific training for licensed personnel in the SER. In FSAR Amendments 8 and 9, the applicant has identified site-specific training contained in FSAR Appendices 13 I-K and has provided adequate responses to RAIs 630.4 and 630.6. The staff has reviewed this information and has determined that the training is appropriate for shift technical advisors. Therefore, this item is resolved.

In FSAR Amendment 9, the applicant has added additional segments to the site fire brigade training. The staff has reviewed the revised fire brigade training program and concludes that the training provided now meets the guidelines in Branch Technical Position CMEB 9.5-1.

15 SAFETY ANALYSIS

15.6 Decrease in Reactor Coolant Inventory

15.6.5 Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Containment Leakage Contribution

As discussed in Section 6.2.3 of this supplement, the applicant has requested the staff's approval in revising the secondary containment drawdown from 168 sec to 375 sec for use in the LOCA analysis. The staff has performed a new LOCA radiological consequence analysis using this 375-sec drawdown time. The calculated doses resulting from the LOCA are summarized in revised Table 15.1. The assumptions used in calculating the design-basis LOCA doses are summarized in revised Table 15.5.

On the basis of this analysis, the staff concludes that use of the 375-sec drawdown time does not change the conclusion previously presented in SER Section 15.6.5.4. Specifically, the radiological consequences of such an accident will be within the exposure guidelines set forth in 10 CFR 100.11.

15.9 TMI Action Plan Requirements

15.9.3 II.K.3 Final Recommendations of Bulletins and Orders Task Force

II.K.3.18 Modification of ADS Logic

In FSAR Amendment 5, the applicant adopted the results of the BWR Owners Group (BWROG) report ("NUREG-0737, Item II.K.3.18--Modification of Automatic Depressurization Systems (ADS) Logic--Feasibility for Increased Diversity for Some Events," dated October 28, 1982) on TMI Action Plan Item II.K.3.18. The applicant has committed to modify the ADS logic to bypass the high drywell pressure trip after a sustained lower water level signal and to add a manual switch that may be used to inhibit ADS actuation if necessary. This is consistent with Option 4 of the BWROG study and is acceptable to the staff.

In the SER, the staff specified the following four conditions that had to be satisfied:

(1) Installation must be completed before initial criticality.

(2) Technical Specifications must be provided for the bypass timer and manual inhibit switch.

(3) The use of the inhibit switch must be addressed in the plant emergency procedures.

(4) A plant-specific analysis must be provided to justify the bypass timer setting.

Hope Creek draft Technical Specifications include the bypass timer and the manual inhibit switch. The use of the manual inhibit switch is addressed in plant operating procedures OP.EO.ZZ-101(Q), "Reactor/Pressure Vessel (RPV) Control," and OP-EO.ZZ-201(Q), "Level Restoration." Plant-specific analyses were completed to justify the bypass-timer setting. The applicant has satisfied Conditions 2, 3, and 4 above. The staff will verify that the installation of the modifications is completed before initial criticality. This confirmatory issue is considered resolved.

II.K.3.31 Plant-Specific Calculations To Show Compliance With 10 CFR 50.46

The staff's review of the GE emergency core cooling system model for NUREG-0737, Item II.K.3.30, found the existing GE small-break LOCA model in continued compliance with 10 CFR 50, Appendix K. Therefore, plant-specific analyses other than those already submitted and approved need not be submitted to satisfy Item II.K.3.31 (see Section 6.3.5 of this supplement). Accordingly, no further action is required by the applicant to satisfy TMI Action Plan Item II.K.3.31.

Table 15.1 Radiological consequences of design-basis accidents (revised from SER)

	Exclusion area* 2-hour dose (rem)		Low population zone** 8-hour dose (rem)†	
Postulated accident	Thyroid	Whole body	Thyroid	Whole body
Main steamline failure outside containment				
With concomitant iodine spike	1.1	<1	<1	<1
With preaccident iodine spike	22	<1	<1	<1
Rod drop accident	<1	<1	<1	<1
Fuel handling accident	5.4	<1	<1	<1
Loss-of-coolant accident (LOCA), duration				
<u>Containment leakage</u>				
0.0 to 2.0 hr	77	1	4	<1
2.0 to 8.0 hr	-	-	<1	<1
8.0 to 24.0 hr	-	-	<1	<1
24.0 to 96.0 hr	-	-	<1	<1
96.0 to 720.0 hr	-	-	<1	<1
<u>Emergency core cooling system leakage</u>				
0.0 to 2.0 hr	47	<1	2	<1
2.0 to 720.0 hr	-	-	<1	<1
<u>Main steam isolation valve leakage</u>				
0.0 to 2.0 hr	-	-	-	-
2.0 to 720.0 hr	-	-	-	-
Total LOCA doses	124	1	6	<1

*Exclusion area boundary (EAB) distance = 901 m.
**Low population zone (LPZ) boundary distance = 8,045 m.
†The calculated LPZ doses after 8 hours for the above accidents other than LOCA were determined to be negligible.

Table 15.5 Assumptions used to evaluate the loss-of-coolant accident (revised from SER)

Parameter	Value
Power level, MWt	3,458
Operating time, years	3
Core fraction airborne in the drywell, %	
Noble gases	100
Iodines	25
Primary containment leak rate, %/day	1.0
Containment free volume, ft^3	301,000
Reactor enclosure free volume, ft^3	4,000,000
Reactor enclosure mixing fraction, %	50
Reactor enclosure building drawdown time, sec	378
Reactor enclosure filtered recirculation and ventilation system exhaust rates, ft^3/min	
0-378 sec	9,000
378 sec to end of accident (720 hours)	336
Recirculation system flow rate, ft^3/min	120,000
Ventilation system iodine filter efficiencies, %	
Elemental	99
Organic	99
Particulate	99
Recirculation system iodine filter efficiencies, %	
Elemental	95
Organic	95
Particulate	95
Minimum exclusion area boundary distance, m	901
Low population zone distance, m	8,045

APPENDIX A

CONTINUATION OF CHRONOLOGY

February 11, 1985	Letter from applicant forwarding Amendment 9 to Final Safety Analysis Report (FSAR) for Hope Creek Generating Station, consisting of text changes due to resolution of SER open items, updated drawings tabulated in Section 1.7, and general revisions.
February 11, 1985	Letter to applicant advising that August 20, 1984, response to draft SER Open Item 103 is acceptable with exception of certain items regarding seismic and dynamic qualification and pump and valve operability assurance.
February 13, 1985	Summary of December 18, 1984, meeting with applicant, National Bureau of Standards, A.D. Little, Inc, and Dames & Moore regarding riverborne missiles.
February 14, 1985	Letter from applicant forwarding response to October 23, 1984, request for additional information on SER Open Item 3 regarding preservice inspection program, based on November 26, 1984, meeting with staff.
February 15, 1985	Letter from applicant forwarding current list regarding status of open and confirmatory items identified in SER Sections 1.7 and 1.8.
February 19, 1985	Letter to applicant informing of March 11-22, 1985, materials independent verification inspection. Construction activities and materials will be sampled through nondestructive examinations as supplemental to existing inspection efforts. Documentation packages for welds requested by February 25, 1985.
February 20, 1985	Letter from applicant forwarding updated master listing of seismic and dynamic qualification summary and status of safety-related equipment, per December 11, 1984, request. New schedule for onsite audit of equipment because of sufficient progress of program requested.
February 22, 1985	Letter from applicant forwarding responses to Questions 5, 6, 7, 8, 20, and 24 regarding potential riverborne missiles. Detailed probability analyses required to respond to Question 3 are still in progress and will be submitted by March 20, 1985.
February 26, 1985	Summary of January 22, 1985, meeting with applicant and Essex Corp. in Bethesda, Maryland, regarding safety parameter display system.

February 27, 1985	Letter from applicant forwarding tabulation of FSAR commitments for January 1985 and resolution for each item. Responses to FSAR sections and TMI Items I.A.3.1, II.B.4, and II.K.3 27 also enclosed. Information will be included in FSAR Amendment 10.
February 27, 1985	Summary of February 12, 1985, meeting with applicant, GE, and Bechtel in Bethesda, Maryland, regarding Technical Specification development and review process.
February 28, 1985	Letter from applicant forwarding response to Item 3 of staff's October 21, 1984, request for additional information regarding performance testing of BWR safety/relief valves for equipment qualification.
February 28, 1985	Letter from applicant forwarding Revision 2 to Section 2.1, "Equipment Classification and Vendor Interface," in response to December 17, 1984, request. Revision provides additional information on master equipment list.
February 28, 1985	Summary of February 12, 1985, meeting with GE, Sargent & Lundy, and Bechtel in Bethesda, Maryland, regarding independent design verification program (IDVP).
March 1, 1985	Letter from applicant forwarding status of open and confirmatory items identified in SER Sections 1.7 and 1.8. Resolutions of listed SER items also enclosed for review and approval.
March 1, 1985	Letter to applicant forwarding interim technical position regarding preservice inspection of pipe welds with corrosion-resistant cladding. Specific information to support final conclusions identified.
March 7, 1985	Letter to applicant forwarding request for additional information regarding quality assurance (QA) program to determine whether Amendments 8 and 9 to FSAR degrade QA program.
March 7, 1985	Letter from applicant forwarding Revision 0 to Test Report T-10656, "Test Report for 20 KVA Uninterruptible Power Supply System Power Circuit Isolation Test," regarding use of inverter as isolation per SER Confirmatory Item 30.
March 8, 1985	Letter from applicant requesting written confirmation of activities concerning exercise of emergency plan leading to issuance of full-power license for Hope Creek. Two onsite exercises for Salem and Hope Creek control rooms will be conducted in 1985.
March 12, 1985	Letter from applicant advising that augmented inservice inspection program for outboard feedwater check valves

	will be provided 6 months after commercial operation. Commitment sufficient to close out SER Outstanding Issue 4.
March 19, 1985	Letter from applicant forwarding revised pages to preservice examination plan originally submitted on April 15, 1984, in response to SER Open Item 3.
March 19, 1985	Letter to applicant forwarding request for additional information regarding responses to Generic Letter 83-28, Items 2.1, 2.2.1, 2.2.2, 3.1.3, 3.2.3, and 4.5.3.
March 22, 1985	Letter from applicant forwarding Revision 2 to draft Technical Specifications, comprised of radioactive effluents and radiological environmental monitoring.
March 26, 1985	Letter to applicant discussing April 15, 1985, response to Generic Letter 82-33 (Supplement 1 to NUREG-0737). Applicant's justification for exceptions to Regulatory Guide 1.97 for some items acceptable.
March 27, 1985	Letter from applicant forwarding additional information on TMI Action Item II.K.3.28, "Qualification of Accumulators on Automatic Depressurization System Valves," per January 25, 1985, request.
March 29, 1985	Letter from applicant forwarding revised "Program Plan for Hope Creek Independent Design Verification Program" and summary of engineering and design program for review.
April 4, 1985	Letter from applicant forwarding Revision 1 to "Program Plan for Hope Creek Independent Design Verification Program," per March 29, 1984, request.
April 4, 1985	Letter from applicant forwarding FSAR commitment status through March 1985, documenting status of utility's responses to NRC requests for additional information.
April 4, 1985	Letter from applicant forwarding proprietary revised FSAR Sections 1.8.1.75 and 8.1.4.14.3 and 17730-01, "Test Report on Cable and Raceway Physical Separation...," in response to SER Open Item 7 and Confirmatory Item 29.
April 4, 1985	Letter from applicant forwarding Revision 7 to emergency plan.
April 5, 1985	Letter to applicant informing that pump and valve operability and seismic qualification audits will be conducted during week of May 6, 1985. Qualification forms for equipment listed in Enclosures 1 and 2 requested by April 22, 1985.
April 8, 1985	Letter from applicant forwarding solid waste process control program for Class A wastes, per SER Section 11.4.2

	and 10 CFR 61. Program to determine stability of Class B and C wastes scheduled for completion by January 1986.
April 10, 1985	Letter from applicant forwarding Revision 0 to "Plant-Specific Technical Guidelines" for preparation of emergency operating procedures.
April 10, 1985	Letter from applicant forwarding Revision 1 to "Safety Analysis for Hope Creek Generating Station SPDS - Display Design and Implementation," per SER Outstanding Issue 15.
April 10, 1985	Letter from applicant forwarding current status of Technical Specification issues identified in Section 16 of SER.
April 11, 1985	Letter to applicant commenting on applicant's April 4, 1985, submittal of Revision 1 to IDVP plan. Revision 1 acceptable, subject to agreement with listed NRC comments and interpretations.
April 11, 1985	Summary of March 29, 1985, meeting with applicant, Sargent & Lundy, Conner & Wetterhahn, Westec, and Kerr-McGee in Bethesda, Maryland, regarding Revision 0 to IDVP plan.
April 15, 1985	Letter from applicant confirming discussions with NRC regarding Sargent & Lundy participation in a research effort concerning cycling operation of utility's fossil-fired unit.
April 16, 1985	Generic Letter 85-06 to all PWR licensees and all applicants for operating licenses regarding QA guidance for anticipated transient without scram equipment not safety related.
April 22, 1985	Letter from applicant responding to SER Confirmatory Issue 13 regarding commitment to provide piping cross-tie on feedwater line fill network.
April 22, 1985	Letter from applicant forwarding current status of open and confirmatory items identified in Sections 1.7 and 1.8 of SER and resolution of SER Item C-37.
April 23, 1985	Letter from applicant forwarding long equipment qualification forms for Pump and Valve Operability Review Team (PVORT) and Seismic Qualification Review Team (SQRT) audit components.
April 24, 1985	Letter from applicant responding to December 21, 1984, letter transmitting Advisory Committee on Reactor Safeguards report on facility.
April 25, 1985	Letter from applicant forwarding 31 oversize piping and instrumentation diagrams and equipment location drawings

	for nuclear steam supply system PVORT and SQRT audit items.
April 26, 1985	Letter from applicant forwarding Revision 0 to "Offsite Dose Calculation Manual."
April 26, 1985	Letter from applicant forwarding Revision 2 to Training Procedure TP-305HC, "NRC-Licensed Operator Requalification Program," per SER Open Item 12(b).
May 1, 1985	Letter from applicant requesting NRC approval to use 100%/day inleakage rate and associated drawdown time as acceptance criteria for preoperational and periodic testing of filtration recirculation ventilation system.
May 1, 1985	Letter from applicant forwarding Amendment 10 to FSAR for Hope Creek Generating Station.
May 1, 1985	Letter from applicant forwarding "Environmental Protection Plan (Nonradiological)."
May 2, 1985	Generic Letter 85-07 to all operating reactor licensees concerning implementation of integrated schedules for plant modifications.
May 8, 1985	Letter from applicant forwarding response to Question 3 on the staff's September 17, 1984, request for additional information regarding potential riverborne missiles and "Annual Probabilities of Exceedance of Significant Dam Deformation."
May 9, 1985	Letter from applicant requesting approval to apply Code Case N-411 to piping as-built reconciliation and selected portions of 1981 Winter Addenda to ASME Code, Section III, for pipe support evaluations.
May 9, 1985	Letter to applicant regarding February 1, 1985, request to use NB-2510 of Section III, 1983 Edition, Summer 1983 Addenda of the ASME Code.
May 13, 1985	Letter from applicant forwarding status of Technical Specification issues from SER Section 16.
May 14, 1985	Letter to applicant requesting additional information regarding the procedures generation package.
May 14, 1985	Letter from applicant forwarding response to staff request for additional information regarding Regulatory Guide 1.97.
May 15, 1985	Letter to applicant requesting additional information regarding the solid waste process control program.

May 16, 1985	Letter to applicant advising of Caseload Management Team visit to site on June 26 and 27, 1985.
May 17, 1985	Letter from applicant forwarding list of FSAR commitments through April 1985. Commitment resolutions provided.
May 21, 1985	Letter from applicant forwarding information in response to 10 CFR 50.62.
May 21, 1985	Letter from applicant responding to staff's March 19, 1985, request for additional information regarding Generic Letter 83-28.
May 22, 1985	Letter from applicant notifying staff of plan to load fuel on December 1, 1985.
May 23, 1985	Letter from applicant responding to staff's March 7, 1985, request for additional information regarding the QA program.
May 23, 1985	Letter from applicant responding to staff's April 5, 1985, request for additional information regarding the Special Nuclear Material License.
May 24, 1985	Letter from applicant forwarding current status of SER open and confirmatory items. Resolution of certain SER items included.
May 30, 1985	Letter from applicant forwarding summary of onsite meetings with NRC staff on May 16 and 17, 1985, regarding emergency preparedness.
May 30, 1985	Letter from applicant forwarding public version of Revision 0 to the Emergency Event Classifications Guide.
May 30, 1985	Letter to applicant identifying open and confirmatory items generated by the May 7-10, 1985, equipment qualification audits.
June 5, 1985	Letter from applicant forwarding test plan for Class 1E digital systems per SER Open Item 5.
June 6, 1985	Letter from applicant forwarding sample environmental qualification checklist and Revision 2 to "Environmental Design Criteria for Hope Creek Generating Station."
June 6, 1985	Summary of April 25, 1985, diesel generator/bus duct configuration meeting.
June 7, 1985	Letter from applicant requesting approval to apply Code Case N-413 to piping as-built reconciliation program.
June 10, 1985	Letter from applicant forwarding status of responses to FSAR commitments to be responded to by May 1985.

June 10, 1985	Letter from applicant forwarding status list of Technical Specification issues. Revision 5 to Technical Specifications also included.
June 11, 1985	Letter from applicant requesting approval to eliminate postulation of intermediate arbitrary pipe breaks.
June 11, 1985	Letter from applicant forwarding information in response to SQRT Confirmatory Issue 3 and equipment-specific Open Item B.1.
June 11, 1985	Letter from applicant forwarding marked-up FSAR changes to be incorporated in Amendment 11 per SER Open Item 2.
June 12, 1985	Letter from applicant providing information regarding SER Confirmatory Item 14.
June 12, 1985	Letter from applicant providing revised response to staff's November 16, 1984, request for additional information regarding the plant-unique analysis report.
June 13, 1985	Letter from applicant updating information regarding the control of heavy loads.

APPENDIX B

BIBLIOGRAPHY

BWR Owners Group, "NUREG-0737, Item II.K.3.18--Modification of Automatic Depressurization Systems (ADS) Logic--Feasibility for Increased Diversity for Some Events," Oct. 28, 1982.

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C. (includes general design criteria).

Denton, H. R. (NRC), letter to all power reactor applicants and licensees, "Qualification of Reactor Operators," Mar. 28, 1980.

General Electric Co., Topical Report NEDE-21175-3, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss of Coolant (LOCA) Loadings," Nov. 1976.

Thomas, C. O. (NRC) letter to J. F. Quirk (GE), "Forwards Safety Evaluation of GE Topical Report NEDE-21175-3, 'BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings, Amendment 3,'" Oct. 20, 1983.

U.S. Nuclear Regulatory Commission, Generic Letter 82-33, "Supplement 1 to NUREG-0737--Requirements for Emergency Response Capability," Dec. 17, 1982.

---, Generic Letter 84-10, "Administration of Operating Tests Prior to Initial Criticality," Apr. 26, 1984.

---, Generic Letter 84-16, "Adequacy of On-Shift Operating Experience for Near Term Operating License Applicants," June 27, 1984.

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

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980; Supp. 1, Jan. 1983 (also issued as Generic Letter 82-33).

---, 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-1021, "Operator Licensing Examiner's Standards," Oct. 1983; Rev. 1, Feb. 1985.

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

APPENDIX D

ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BWR	boiling-water reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
FSAR	Final Safety Analysis Report
GDC	general design criterion(a)
GE	General Electric Company
HTFFT	heat transfer, fluid flow, and thermodynamics
IDVP	independent design verification program
IE	Office of Inspection and Enforcement
ISI	inservice inspection
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
NRC	U.S. Nuclear Regulatory Commission
PSE&G	Public Service Electric and Gas Company
PVORT	Pump and Valve Operability Review Team
PWR	pressurized-water reactor
QA	quality assurance
RAI	request for additional information
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RG	regulatory guide
RHR	residual heat removal
RPM	Radiation Protection Manager
SACS	safety auxiliary cooling system
SER	Safety Evaluation Report
SPDS	safety parameter display system
SQRT	Seismic Qualification Review Team
SSWS	station service water system

TMI	Three Mile Island
TSSP	technical supervisory skills program
WG	water gauge

APPENDIX E

PRINCIPAL STAFF CONTRIBUTORS AND CONSULTANTS

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

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APPENDIX K

SAFETY EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING

SAFETY EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.D.1)
RELIEF AND SAFETY VALVE TESTING
FOR THE HOPE CREEK GENERATING STATION
OF THE PUBLIC SERVICE
ELECTRIC AND GAS COMPANY
DOCKET NO. 50-354

1. INTRODUCTION

1.1 Background

Light water reactor experience has included a number of instances of improper performance of relief and safety valves installed in the primary coolant systems. There have been instances of valves opening below set pressure, valves opening above set pressure and valves failing to open or reseat. From these past instances of improper valve performance, it is not known whether they occurred because of a limited qualification of the valve or because of a basic unreliability of the valve design. It is known that the failure of a power-operated relief valve to reseat was a significant contributor to the TMI-2 sequence of events; however, such an event in a Boiling Water Reactor (BWR) would not have the same severe consequences. Nevertheless, these facts led the task force which prepared NUREG-0578[1] to recommend that programs be developed and executed which would reexamine the performance capabilities of BWR safety and relief valves for unusual but credible events. These programs were deemed necessary to reconfirm that the General Design Criteria 14, 15 and 30 of Appendix A to Part 50 of the Code of Federal Regulations, 10 CFR are indeed satisfied.

1.2 General Design Criteria and NUREG Requirements

General Design Criteria 14, 15, and 30 require that (1) the reactor primary coolant pressure boundary be designed, fabricated and tested so as

to have an extremely low probability of abnormal leakage, (2) the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation or anticipated transient events and (3) the components which are part of the reactor coolant pressure boundary shall be constructed to the highest quality standards practical.

To reconfirm the integrity of relief and safety valve systems and thereby assure that the General Design Criteria are met, the NUREG-0578 position was issued as a requirement in a letter dated September 13, 1979 by the Division of Licensing (DL), Office of Nuclear Reactor Regulation (NRR) to ALL OPERATING NUCLEAR POWER PLANTS. This requirement has since been incorporated as Item II.D.1 of NUREG-0737[2] (Clarification of TMI Action Plan Requirements) which was issued for implementation on October 31, 1980. As stated in the NUREG reports, each boiling water reactor Licensee and applicant shall:

1. Conduct testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

2. Determine valve expected operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2.

3. Choose the single failures such that the dynamic forces on the safety relief valves are maximized.

4. Use the highest test pressures predicted by conventional safety analysis procedures.

5. Include in the relief and safety valve qualification program the qualification of the associated control circuitry, piping and supports.

6. Test data including criteria for success or failure of valves tested must be provided for Nuclear Regulatory Commission (NRC) staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

7. Each Licensee must submit a correlation or other evidence to substantiate that the valves tested in a generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report (FSAR). The effect of as-built relief and safety valve discharge piping on valve operability must be accounted for if it is different from the generic test loop piping.

2. BWR OWNERS' GROUP RELIEF AND SAFETY VALVE PROGRAM

To respond to the NUREG requirements listed above, the BWR Owners' Group contracted the General Electric Company (GE) to design and conduct a Safety/Relief Valve Test Program.[3] The program describes the safety/relief valves to be tested, the test facility requirements, the test sequence, the valve acceptance criteria and the procedure for obtaining, analyzing and reporting the test data. Prior to its acceptance, the test program received extensive NRC review and comment followed by responses from the GE/BWR Owners' Group. Six NRC questions and Owners' Group responses dealing with justification of the applicability of test results to the in-plant safety/relief valves are contained in the enclosure to Reference 4. The NRC review of the response to these questions is contained in Reference 5. Based on this review, the concerns expressed in the questions were appropriately resolved.

The early BWRs contain a combination of dual function safety/relief valves (SRV), power actuated relief valves (PARV) and single function safety valves (SV). At the Hope Creek Generating Station, there are 14 two-stage, dual function SRV's. There are no PARV's or single function SV's at the Hope Creek Generating Station.

The qualification of the SRVs for steam discharge under expected operating and accident conditions has been demonstrated by vendor production tests and is confirmed routinely by in-plant startup and operability tests. Based on this, it was agreed that the valves should be tested for those events that result in liquid or two-phase flow at the SRV.

The test sequence and conditions established in the test program were based on an evaluation of expected operating conditions determined through the use of analyses of accident and anticipated operational occurrences referenced in Regulatory Guide 1.70, Rev. 2. Enclosure 2 to Reference 3 provides this evaluation which indicated that there is one event which is significantly likely to occur and can lead to the discharge of liquid or two-phase flow from the SRVs. This event combined with the single failure requirement of NUREG 0737 results in the conclusion that a test should be

performed simulating the alternate shutdown cooling mode which utilizes the SRVs as a return flow path for low pressure liquid to the suppression pool.

At a meeting on March 10, 1981,[6] the BWR Owners' Group presented results of a study by Science Applications, Inc. (SAI) which showed that the probability of getting liquid to the steam line, and hence to the SRV's is approximately 10^{-2} per reactor year. However, even if the water level increases to the mid-plane of the steam line nozzle on the vessel, which is not likely,[a] the fluid quality at the valve was calculated by GE to be greater than 20%.[3] Because the steam lines typically drop about 45 feet vertically from the vessel nozzles to the horizontal runs on which the SRVs are mounted, much of the liquid which gets to the steam lines would be entrained as droplets. Therefore, the two-phase mixture upstream of the SRVs, should liquid reach the level of the steam lines, would exist as a froth, droplet, annular or stratified flow regime, and slug flow or subcooled liquid flow would be unlikely.

Even if two-phase discharge through a SRV should result in a stuck open valve, the results of the blowdown are not severe. As discussed in Reference 7, historically there have been a total of 53 inadvertent blowdown events due to pressure relief system valve malfunctions from 1969 through April 1978. These events varied in consequences from a short duration pressure transient to a rapid depressurization and cooldown of the primary coolant system from approximately 1100 psig to a few hundred psig. No fuel failures due to these transients have been reported.

In Reference 8, the BWR Owners' Group discusses the consequences of the worst case transient for maintaining the core covered (loss of feedwater) combined with the worst single failure (failure of the high pressure injection system) and one stuck open relief valve. Reference plant analyses for a BWR/4 and a BWR/5 show that the Reactor Core Isolation

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a. Feedwater pumps would be tripped prior to the water level reaching the mid-plane by the L8 high level trip, turbine vibration trip, or by operator action.

Cooling (RCIC) system can automatically provide sufficient inventory to keep the core covered. This capability is not a design basis for the RCIC system and not all plants have been analyzed to demonstrate this capability. If a plant should not have this capability, manual depressurization to low pressure core cooling systems will avoid core uncovery for the case of loss of feedwater plus worst single failure plus a stuck open relief valve. Therefore, even for the loss of feedwater transient with the worst single failure, a stuck open relief valve does not uncover fuel.

At the March 10, 1981 meeting,[6] the BWR Owners' Group presented an analysis that showed that even if a slug of subcooled water exists upstream of the SRVs, the probability of rupturing the discharge line is 7×10^{-4} per event. The Staff has not reviewed the supporting analysis for this value; however, even if the failure probability is as high as 10^{-2} per event, the combined probability is no greater than for a steam line break inside containment. GE states that the steam line break, which has been analyzed and found to be acceptable, would be more severe (effects on the core and containment) than a break in a SRV discharge line with a stuck open SRV because the assumed break area is larger.

In summary, based on the BWR operating history of inadvertent SRV blowdowns, the low likelihood of severe consequences, and the bounding design basis steam line break, the staff decided not to require high pressure testing with saturated liquid or subcooled water.

Based on the above, the Licensee has complied with NUREG Requirements 1-4 (Paragraph 1.2 above). That is, an acceptable test program was established which adhered to the Staff guidelines on the selection of test conditions and the maximization of system loads. That portion of Item 5 dealing with the qualification of the associated control circuitry is considered to be satisfied as a result of the anticipated licensing action for compliance with 10 CFR, Part 50.49.

3. BWR OWNERS' GROUP TEST RESULT AND ANALYSIS

In October 1981, the BWR Owners' Group published a technical report[9] documenting the results of the prototypical safety/relief valve tests conducted in accordance with the accepted Test Program.[3] The tests were performed by the General Electric Company for the BWR Owners' Group at the Wyle Laboratory in Huntsville, Alabama. The test report, which was reviewed by the Staff, describes the test facility, the basis for the test conditions and valve selection, the instrumentation and its accuracy, and analyzes the results with respect to valve operability, piping and support loads and the applicability of the test results to the in-plant safety and relief valves.

With the completion of the testing and the submittal of the test report, the Licensees complied with NUREG Requirement No. 6 listed in 1.2 above. However, the subsequent Staff review of the test results generated four plant specific questions stated in Reference 10 which required resolution. References 11 and 12, representing the Hope Creek Generating Station responses to the four plant specific questions, were submitted for review in February, 1985.

4. REVIEW AND EVALUATION

4.1 Review of Test Results and Analysis

An extensive review[13,14] of the test results[9] was conducted by NRC consultants (EG&G Idaho, Inc.) at the Idaho National Engineering Laboratory. The review addressed not only the test results but also the applicability of the test results and equipment to the Hope Creek safety-relief valve systems. The four plant specific questions generated by the review and the Licensee responses to those questions are discussed in Paragraph 4.4 below.

4.2 Valves Tested

The generic test program required the testing of six different safety/relief valves. Included was a Target Rock 6 x 10 Two-Stage Horizontal Discharge Pilot Operated Safety/Relief Valve, Model No. 7567F. This valve, with minor differences, is the valve used at the Hope Creek Generating Station. The tested valve was different from the plant valves in the following areas:

1. Topwork design

2. Seat bore diameter

3. Main disc lift position

4. Valve Orientation.

The only differences in the top works are dimensional which would not affect the operability of the valve or the piping reaction loads from water discharge. Exact dimensions for the Hope Creek valves were not provided in the test report, however the Owners' Group in-plant valves have seat bore diameters and disc lift values that range from 4.27 in. and 2.58 in. respectively to 5.125 in. and 2.63 in. respectively. The two-stage Target Rock test valve has a 5.125 in. diameter seat bore and a 2.63 in. lift, thereby bounding the maximum flow capacity.

Although the Hope Creek Generating Station does not employ the Three-Stage Target Rock valve, it was also included in the test program. The three-stage test valve has a bore diameter of 4.27 in. and was considered bounding from an operational standpoint since flashing under the water test conditions would be more likely to occur with the smallest bore diameter. Thus, the two-stage test valve bounds the maximum flow capacity and discharge line loads that could be expected for the in-plant valves, and the three-stage test valve verified the operability of the Hope Creek Generating Station in-plant valves.

4.3 Test Conditions

As discussed in Section 2.0 herein, test conditions to envelop the expected BWR Safety/Relief Valve events were developed in accordance with NRC guidelines. They were accepted and are presented in Reference 3. The review of the test results indicates that the actual test conditions were in accordance with the established test program.

4.4 Evaluation of Responses to Plant Specific Questions

The response to Question No. 1 indicates that there are valve discharge line differences between the test configuration and the in-plant configuration. However, it is pointed out that these differences result in bounding loads on the SRV's. The first segment of test piping downstream of the safety valve is longer than the comparable in-plant segment (12 ft vs 10 ft-2 in.) which would result in a higher moment at the test valve. Discharge from the tee quencher at the end of the Hope Creek Generating Station SRV discharge line cannot transmit loads to the valve as the test system could because the in-plant line is anchored between the quencher and the valve. Thus, this portion of the response (dynamic, mechanical loads) is considered to be acceptable. The second part of the response addressed the back pressure (dynamic, hydraulic) loads on the test and in-plant valves. The Licensee addressed both transient and steady state back-pressure loads. The steady state back pressure for the test valve was forced to be greater than that expected in-plant by installing a predetermined orifice plate in the discharge line before the ram's head and above the water line. The response also indicated that the high pressure steam test preceding the low pressure water test would produce the greater transient back pressures between the two tests. This would be true due to the higher pressure upstream of the SRV and the shorter valve opening time. The discharge line average length at the Hope Creek Generating Station is greater than in the test facility (125 ft vs 112 ft) and the submergence length at the plant is less than in the test facility (7-10 ft vs 13 ft), both of which result in greater back pressure in test facility.

Based on the above discussion, the response to the first question is considered by the Staff to be acceptable.

The response to the second question described the support system components in the Hope Creek Generating Station discharge lines indicating that spring hangers do exist at Hope Creek whereas the test facility piping did not include spring hangers. The basic argument defending the adequacy of the spring hangers (in fact, all supports) is that they were designed for the

much larger, high steam pressure relief valve opening loads. In this case, therefore, sufficient margin is available in the in-plant spring hangers to account for the additional load due to the dead weight in the water-filled, low pressure event. The test results indicated significantly lower dynamic loads during the water discharge event than during the high pressure steam discharge case and the point made in this response (as well as in the response to Question No. 1) is that the test program was designed primarily to demonstrate valve and system adequacy under the prototypical water discharge events (i.e., the alternate shutdown cooling mode).

Thus, with the in-plant safety/relief valve discharge piping and support system designed for the high pressure steam discharge event and with the satisfactory response of the test valves, the discharge piping and support system to the low pressure water blowdown, the reply to the second question is considered by the Staff to be acceptable.

Question No. 3 asked the Licensee to describe and compare expected events at the Hope Creek Generating Station with the test conditions of the generic test program. The Licensee summarizes the analysis procedure(3) using Regulatory Guide 1.70 which arrived at 13 events that would result in liquid or two-phase flow through the SRV's and maximize the dynamic forces on the valve. As indicated in Section 2.0 herein, this analysis concluded that the alternate shutdown cooling mode is the only expected event which will result in liquid at the valve inlet. To simulate this event the test program(3) used a 15-50°F subcooled liquid at 20-250 psig at the SRV inlet prior to valve opening. The Licensee indicates that the alternate cooling mode of operation at the Hope Creek Generating Station will result in subcooled fluid (Approximately 20°F) at a pressure of approximately 50 psig. Therefore, the test conditions envelop the expected conditions for this event should it occur at the Hope Creek Generating Station. The Licensee's response to the third question is acceptable to the staff.

The response to the forth question addresses the determination and future use of the valve flow coefficient, C_v. The response indicates that the value of the liquid flow coefficient, in itself, is not of direct

interest. The flow capacity of the valves as measured during the tests is the data of interest. The flow capacity of the system SRV's is larger than the capacity of the coolant source pump of the residual heat removal (RHR) system and therefore sufficient to remove decay heat. The answer to this question is considered to be acceptable to the Staff.

Considering the above evaluations, the Staff finds that the Licensee for the Hope Creek Generating Station has provided an acceptable response to NUREG Item 7 and to the piping and support concerns of NUREG Item 5 (Paragraph 1.2 herein).

4.5 Supporting Information

4.5.1 Additional Questions

Two other questions generated by the staff concerning (1) valve functional deficiencies encountered during invalidated test runs and (2) the effect of steam cycling on valve performance have been addressed previously by other Licensees using the Target Rock Two-Stage SRV. The staff accepted those responses based on the following:

1. Previous submittals by other Licensees have stated, "All the valves subjected to test runs, valid or invalid, opened and closed without loss of pressure integrity or damage." This statement was supported by the Wyle Laboratory test log sheet for the Target Rock valve.

2. Although the test program did not subject the valves to steam cycling, the valve vendor has subjected his valves to high pressure steam flow cycling and no loss of valve performance has been noted.

Because of this prior acceptance, the Licensee for the Hope Creek Generating Station, Units 1 and 2, was not requested to respond to these concerns.

4.5.2 High Pressure Steam Flow/Discharge Piping Response

The applicability of the response of the safety-relief valve discharge piping system to the response of the in-plant piping system has been accepted above. In the test report,[9] it is indicated that, (1) the analytically predicted response of the test piping and supports was comparable to the measured values, and (2) the maximum test piping response to liquid flow was generally less than 30% of that due to test steam flow conditions. Further, as part of the initial review, the loads on the in-plant piping and supports due to steam discharge were found to be acceptable by the Staff. It should also be mentioned that the staff's on-going review of the Mark-I Containment Long Term Program includes a review of the methods of analysis, computer code adequacy and design criteria for SRV discharge piping supports for high pressure steam discharge conditions.

5. EVALUATION SUMMARY

The Licensee for the Hope Creek Generating Station has provided an acceptable response to the requirements of NUREG-0737, and thereby, reconfirmed that the General Design Criteria 14, 15 and 30 of Appendix A to 10 CFR-50 have been met. The rationale for this conclusion is given below.

The Licensee with concurrence by the Staff developed an acceptable Relief and Safety Valve Test Program designed to qualify the operability of the prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. The subsequent tests were successfully completed under operating conditions which by analysis bounded the most probable maximum forces expected from anticipated design basis events. The generic test results showed that the valves tested functioned correctly and safely for all steam and water discharge events specified in the test program and that the pressure boundary component design criteria were not exceeded. Analysis and review of the test results and the Licensee justifications indicated the direct applicability of the prototypical valve and valve system performances to the in-plant valves and systems intended to be covered by the generic test program.

Thus, the requirements of Item II.D.1 of NUREG 0737 have been met (Items 1-7 in Paragraph 1.2) and, thereby, assure that the reactor primary coolant pressure boundary will have, by testing, a low probability of abnormal leakage (General Design Criterion No. 14) and that the reactor primary coolant pressure boundary and its associated components (piping, valves and supports) have been designed with sufficient margin such that design conditions are not exceeded during relief/safety valve events (General Design Criterion No. 15).

Further, the prototypical tests and the successful performance of the valves and associated components demonstrated that this equipment has been constructed in accordance with high quality standards (General Design Criterion 30).

REFERENCES

1. TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

2. Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

3. Letter, D. B. Waters (BWR Owners' Group), to Richard H. Vollmer (NRC) "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves," September 17, 1980.

4. Letter, D. B. Waters, Chairman BWR Owners' Group to D. G. Eisenhut, Director, Division of Licensing, NRR, USNRC, "Responses to NRC Questions on the BWR S/RV Test Program," BWR06-8135, March 31, 1981.

5. Letter, B. F. Saffell to R. E. Tiller, Comments on BWR Owners' Group Responses to NRC Questions on Safety/Relief Valve Low Pressure Program - Saff-95-81, April 23, 1981.

6. Memorandum to Themis P. Speis from Wayne Hodges, "Summary of March 10 Meeting with General Electric to Discuss BWR Liquid Overfill Events," May 1981.

7. Technical Report on Operating Experience with BWR Pressure Relief Systems, NUREG-0462, July 1978.

8. Letter to Darrell G. Eisenhut (NRC) from David B. Waters (BWR Owners' Group), BWROG-80-12, "BWR Owners' Group Evaluation of NUREG-0737 Requirements," December 29, 1980.

9. Analysis of Generic BWR Safety Relief Valve Operability Test Results, General Electric NEDE-2 4988-P, October 1981.

10. Letter, A. Schwencer (USNRC) to R. L. Mittl (Public Service Electric and Gas Company), November 24, 1984.

11. Letter from R. L. Mittl, Public Service Electric and Gas Company, to A. Schwencer, USNRC, "Equipment Qualification, Hope Creek Generating Station, Docket No. 50-354," February 1, 1985.

12. Letter from R. L. Mittl, Public Service Electric and Gas Company, to A. Schwencer, USNRC, "Equipment Qualification, Hope Creek Generating Station, Docket No. 50-354," February 28, 1985.

13. Letter, B. F. Saffell to R. E. Tiller, "Review of BWR/GE Safety Relief Valve Test Report (A6356)" Saff-14-82, January 13, 1982.

14. Letter, B. F. Saffell to D. E. Solecki, "Open Questions-BWR/GE Safety/Relief Valve Test Report, BWR Owners' Safety/Relief Submittals (A6356)"-Saff-178-82, May 4, 1982.

APPENDIX L

SAFETY EVALUATION REPORT
TMI ACTION--NUREG-0737 (II.E.4.2)
CONTAINMENT PURGE AND VENT
VALVE OPERABILITY

EGG-EA-6861

HOPE CREEK GENERATING STATION
UNIT 1
DOCKET NUMBER 50-354
DEMONSTRATION OF CONTAINMENT PURGE AND VENT VALVE OPERABILITY

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Prepared for the
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Washington, D.C. 20555
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ABSTRACT

The containment purge and vent valve qualification program for the Hope Creek Generating Station has been reviewed by the NRC Licensing Support Section. The review indicates that the licensee has demonstrated the dependability of containment isolation against the buildup of containment pressure due to a LOCA/DBA with the restrictions that (1) during operating conditions 1, 2, and 3 all purge and vent valves will be sealed closed and under administrative control, and (2) during power ascension and descension conditions the 26 in. inboard valve (1-GS-HV-4952) will be used in series with the 2 in. bypass valve (1-GS-HV-4951) to control the release of containment pressure.

FOREWORD

This report is supplied as part of the "Equipment Qualification Case Reviews" project that is being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Engineering, Equipment Qualification Branch by EG&G Idaho, Inc., Engineering Analysis Division, NRC Licensing Support Section.

The U.S. Nuclear Regulatory Commission funded this work under the authorization, B&R 20-19-40-41-2, FIN Number A6415.

SUMMARY

The Hope Creek containment purge and vent valve qualification program has been reviewed by the NRC Licensing Support Section of EG&G Idaho, Inc., who provide technical assistance to the NRC Equipment Qualification Branch (EQB). Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design basis accident, is necessary to assure containment isolation. This demonstration of operability is required by Standard Review Plan (SRP) 6.2.4, Branch Technical Position (BTP) CSB 6-4, and SRP 3.10 for containment purge and vent valves which are not sealed closed during operational conditions 1, 2, 3, and 4.

Material contained in the Hope Creek Final Safety Analysis Report (FSAR) and purge valve qualification report formed the basis for this evaluation. The licensee did not specifically qualify the containment purge and vent valves to LOCA conditions, because either all of the valves will be closed during most plant operating conditions, or only two valves will be opened in a bypass flow configuration that will prevent any LOCA-induced flow through the 26 in. inboard valve from exceeding its normal purge flow. Here, the "normal" purge flow is considered to be flow that will pass through the fully open 26 in. inboard valve during cold shutdown and refueling conditions. In summary, we find that the dependability of containment isolation is demonstrated against the buildup of containment pressure due to a LOCA/DBA, with the restrictions that (1) during operating conditions 1, 2, and 3 all purge and vent valves will be sealed closed and under administrative control, and (2) during power ascension and descension conditions the 26 in. inboard valve (1-GS-HV-4952) will be used in series with the 2 in. bypass valve (1-GS-HV-4951) to control the release of containment pressure.

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HOPE CREEK GENERATING STATION
UNIT 1
DOCKET NUMBER 50-354

DEMONSTRATION OF CONTAINMENT PURGE AND VENT VALVE OPERABILITY

1. REQUIREMENT

Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design basis accident, is necessary to assure containment isolation. This demonstration of operability is required by Standard Review Plan (SRP) 6.2.4, Branch Technical Position (BTP) CSB 6-4, and SRP 3.10 for containment purge and vent valves which are not sealed closed during operational conditions 1, 2, 3, and 4.

2. DESCRIPTION OF CONTAINMENT PURGE AND VENT VALVES

The following valves are identified in the licensee submittal and Hope Creek FSAR Table 6.2-16 as being part of the containment purge and vent valve system.

Valve Tag No.	Size (Inch)	Use	Location
1-GS-HV-4952	26	Drywell Exhaust	Inboard Containment
1-GS-HV-4950	26	Drywell Exhaust	Outboard Containment
1-GS-HV-4951	2	Drywell Exhaust	Outboard Containment (Bypass)
1-GS-HV-4964	24	Suppression Pool Exhaust	Inboard Containment
1-GS-HV-4962	24	Suppression Pool Exhaust	Outboard Containment
1-GS-HV-4963	2	Suppression Pool Exhaust	Outboard Containment (Bypass)

The 24 in. and 26 in. butterfly valves are BIF, air-operated. The 2 in. globe valves are Rockwell, Model 3624 MT, air-operated.

All containment purge and vent valves will be sealed closed and under administrative control during normal plant operating conditions 1, 2, and 3. During power ascension and descension only the 26 in. inboard vent valve (1-GS-HV-4952) and the 2 in. outboard bypass valve (1-GS-HV-4951) will be open to vent the containment as required for thermal expansion and contraction of the air volume. The 26 in. outboard vent valve (1-GS-HV-4950) will remain locked closed. This configuration will ensure that the flow through the inboard 26 in. valve is limited to the choked flow through the 2 in. bypass valve. The choked flow through the 2 in. valve will be less than 20 percent of the cold shutdown purge flow through the 26 in. valve.

3. DEMONSTRATION OF OPERABILITY

The following document was submitted for review by the Public Service Electric and Gas Company to demonstrate dependability of containment isolation regarding operability of the Hope Creek containment purge and vent valves.

1. Letter from R. L. Mittl, General Manager, Nuclear Assurance and Regulation, Public Service Electric and Gas Company to A. Schwencer, Chief, Licensing Branch 2, Division of Licensing, U.S. Nuclear Regulatory Commission, Equipment Qualification, Dependability of Containment Isolation-Containment Purge and Vent Operability, NRC Accession Number 8502050578, February 1, 1985.

The licensee did not specifically qualify the containment purge and vent valves to LOCA conditions, because either all of the valves will be closed during most plant operating conditions, or only two valves will be opened in a bypass flow configuration that will prevent any LOCA-induced flow through the 26 in. inboard valve from exceeding its normal purge flow. Here, the "normal" purge flow is considered to be flow that will pass through the fully open 26 in. inboard valve during cold shutdown and refueling conditions. The following assumptions form the basis for the licensee's submittal.

1. The containment purge and vent valves will be sealed closed and under administrative control during normal plant operating conditions 1, 2, and 3.

2. The 26 in. inboard drywell exhaust valve and the 2 in. bypass valve will be opened in series as required to vent thermal expansion of the air volume during power ascension and descension. The flow rate in this pathway will be less than the normal purge flow of 9000 CFM through the 26 in. valve during cold shutdown and refueling conditions.

3. Containment backpressure does not affect the valve operators, because the valves are located immediately outside of the primary containment.

4. The 2, 24, and 26 inch valves are designed to fail closed and do not depend on air accumulators for fail-safe actuation. Reduced voltage operation is not applicable.

5. The air-operated 2 in. globe valves have handwheels which must be manually disengaged to permit normal operation. Similarly, the hydraulic system which opens the air-operated butterfly valves must be manually disengaged for normal operation. Administrative controls will be used to ensure that the handwheels and hydraulic systems are disengaged following maintenance or related activities prior to returning the valves to normal service.

6. No test was performed to determine the dynamic torque coefficient for LOCA conditions, because the LOCA-induced flow through the 26 in. purge valve will be less than its normal purge flow during cold shutdown and refueling conditions.

The licensee has determined that the operability criteria of SRP 6.2.4, BTP CSB 6-4, and SRP 3.10 are satisfied since the valves will be sealed closed during operational conditions 1, 2, and 3 except to vent the pressure change during power ascension and descension. If a LOCA occurs during either of these operating modes, the exhaust flow rate will be choked by the 2 in. bypass valve and will not exceed 20 percent of the cold shutdown purge flow through the 26 in. valve. Consequently, the ability of the purge and vent valves to provide containment isolation has been demonstrated.

4. EVALUATION

We find that the information submitted by the licensee has demonstrated the ability of the purge and vent valves to provide containment isolation. The following considerations form the basis for our findings.

1. A review of the Hope Creek FSAR Section 6.2.4 shows that during normal operation the 24 in. and 26 in. containment purge valves will be sealed closed except for the inboard valve on the drywell purge outlet vent line, 1-GS-HV-4952. This 26 in. valve in series with the 2 in. outboard bypass valve (1-GS-HV-4951) can be opened periodically to permit venting of the primary containment to relieve pressure. The aerodynamic flow effects on closing torque requirements for the 26 in. valve are minimal, since the maximum flow through the valve will be less than 20 percent of its cold shutdown purge flow.

2. The demonstration of purge valve operability applies to valves larger than 3 in. nominal diameter in accordance with the requirements of NUREG-0737, Section II.E.4.2, Attachment 1. Consequently, the 2 in. bypass valve is exempt from these requirements, although the licensee has indicated in the FSAR 6.2.4.3.2.1 that the 2 in. valve is qualified to close against the LOCA-induced flow.

3. The qualification methodology that was used to address cold shutdown purge flow conditions is outside the scope of this review and was not included in the licensee's submittal. Since any LOCA-induced flow through the 26 in. valve will be less than the normal purge flow during cold shutdown and refueling conditions, the licensee did not provide many qualification details that would usually be considered by the staff to evaluate valve operability. For example, the design ratings of the valves and actuators were not specified. The aerodynamic flow effects of disc shape and piping geometry on valve closing torque

requirements were not discussed. The valve closure was determined by static deflection tests, although the closure times and applied loads were not given. However, in consideration of the operating restrictions imposed upon the valves, as cited in Section 3 Items (1) and (2), plus the operability exemption applied to valves smaller than 3 in., we believe that there is sufficient design margin to conclude that both the 26 in. inboard and 2 in. bypass valves will operate as needed.

4. The licensee's submittal has not demonstrated valve operability for the random presence of debris or damage to the seat. Instead, the containment purge lines are provided with debris screens, the evaluation of which is beyond the scope of this review. Consequently, our evaluation of the containment purge and vent valves does not cover the potentially adverse conditions such as the accumulation of debris, corrosion, entrapment of foreign objects, or damage to the sealing surfaces.

5. CONCLUSIONS

We have completed our review of information concerning the ability of the purge and vent valves to provide containment isolation for the Hope Creek Generating Station Unit 1. In summary, we find that the dependability of containment isolation is demonstrated against the buildup of containment pressure due to a LOCA/DBA, with the restrictions that (1) during operating conditions 1, 2, and 3 all purge and vent valves will be sealed closed and under administrative control, and (2) during power ascension and descension conditions the 26 in. inboard valve will be used in series with the 2 in. bypass valve to control the release of containment pressure.

6. REFERENCES

1. Letter from R. L. Mittl, General Manager, Nuclear Assurance and Regulation, Public Service Electric and Gas Company to A. Schwencer, Chief, Licensing Branch 2, Division of Licensing, U.S. Nuclear Regulatory Commission, Equipment Qualification, Dependability of Containment Isolation-Containment Purge and Vent Operability, NRC Accession Number 8502050578, February 1, 1985.

APPENDIX M

CONFORMANCE TO REGULATORY GUIDE 1.97,
HOPE CREEK GENERATING STATION

CONFORMANCE TO REGULATORY GUIDE 1.97
HOPE CREEK GENERATING STATION

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ABSTRACT

This EG&G Idaho, Inc., report reviews the submittals for Regulatory Guide 1.97, Revision 2, for the Hope Creek Generating Station and identifies areas of nonconformance to the regulatory guide. Exceptions to Regulatory Guide 1.97 are evaluated and those areas where sufficient basis for acceptability is not provided are identified.

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Integration by EG&G Idaho, Inc., NRC Licensing Support Section.

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Docket No. 50-354

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CONFORMANCE TO REGULATORY GUIDE 1.97
HOPE CREEK GENERATING STATION

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

Public Service Electric and Gas Company, the applicant for the Hope Creek Generating Station, provided a response to the generic letter on April 15, 1983 (Reference 4). The letter referred to Sections 1.8 and 7.5 and Table 7.5.1 of the Final Safety Analysis Report (FSAR, Reference 5) for a review of the instrumentation provided for Regulatory Guide 1.97. Additional information was provided by letter dated May 14, 1985 (Reference 6).

This report provides an evaluation of this material.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement No. 1, sets forth the documentation to be submitted in a report to the NRC describing how the applicant complies with Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range

2. Environmental qualification

3. Seismic qualification

4. Quality assurance

5. Redundance and sensor location

6. Power supply

7. Location of display

8. Schedule of installation or upgrade

Furthermore, the submittal should identify deviations from the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March, 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this subject. At these meetings, it was noted that the NRC review would only address exceptions to Regulatory Guide 1.97. Furthermore, where licensees or applicants explicitly state that instrument systems conform to the regulatory guide, it was noted that no further staff review would be necessary. Therefore,

this report only addresses exceptions to Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

3. EVALUATION

The applicant provided a response to NRC Generic Letter 82-33 on April 15, 1983. This response referred to the Final Safety Analysis Report (FSAR) which describes the applicant's position on post-accident monitoring instrumentation. Additional information was provided on May 14, 1985. This evaluation is based on this material.

3.1 Adherence to Regulatory Guide 1.97

Section 7.5.2.3.c of the FSAR states that post-accident monitoring instrumentation is designed and implemented according to Regulatory Guide 1.97, with the clarifications and exceptions as stated in Section 1.8.1.97. This section states that the Regulatory Guide 1.97 requirements are being implemented except where technically justified or where implementation would conflict with another portion of the regulatory guide or some other applicable design criteria. Section 7.5.1.3.4 states that the displayed parameters listed in Table 7.5-1 are furnished in accordance with the criteria of Regulatory Guide 1.97, Revision 2. Therefore, it is concluded that the applicant has provided an explicit commitment on conformance to Regulatory Guide 1.97. Exceptions to and deviations from the regulatory guide are noted in Section 3.3.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required to permit the control room operator to take specific manually controlled safety actions. The applicant classifies the following instrumentation as Type A.

1. Coolant level in reactor

2. Reactor coolant system pressure

3. Containment and drywell hydrogen concentration

4. Containment and drywell oxygen concentration

5. Suppression pool water level

6. Suppression pool water temperature

7. Drywell pressure

This instrumentation meets the Category 1 recommendations consistent with the requirements for Type A variables.

3.3 Exceptions to Regulatory Guide 1.97

The applicant identified deviations and exceptions from Regulatory Guide 1.97. These are discussed in the following paragraphs.

3.3.1 Neutron Flux

Regulatory Guide 1.97 recommends Category 1 instrumentation with a range of from 10^{-6} to 100 percent of full power. The applicant has provided three redundant sets of instrumentation having overlapping ranges which, together, cover the recommended range. However, the instrumentation is Category 2. The applicant states that the source-range monitors and intermediate-range monitors are driven into the core soon after shutdown and this makes it highly probable that one or more of the existing detectors will be inserted. The applicant says that the operator can actuate the standby liquid control system on loss of instrumentation. There are 4 source-range monitors, 8 intermediate-range monitors, 6 average power range monitors and individual linear power range monitors.

This deviation is similar to most boiling water reactors. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 is an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The

applicant has committed to follow industry development of this equipment, evaluate newly developed equipment, and install Category 1 instrumentation when it becomes available.

3.3.2 Drywell Sump Level
Drywell Drains Sump Level

The applicant is supplying instrumentation for this variable that is Category 3 rather than the recommended Category 1. Their justification for this deviation is that

1. The sump level is not an unambiguous indication of a breach in the reactor coolant system pressure boundary

2. Other instrumentation (drywell pressure, drywell temperature and primary containment radiation) indicates leakage in the drywell

3. The sump level does not cause any automatic initiation of safety-related systems or alert the operator to take any safety-related actions

4. The sump level provides only non safety indications

5. The sumps are deliberately isolated at the primary containment penetration upon receipt of an accident signal. This is done to establish containment integrity.

We conclude that the instrumentation supplied by the applicant will provide appropriate monitoring for the parameters of concern. This is based on (a) for small leaks, the instrumentation is not expected to experience harsh environments during operation, (b) for larger leaks, the sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, thus negating the drywell sump level and drywell drain sumps level instrumentation, (c) the drywell pressure and temperature as well as the primary containment area radiation instrumentation can be used

to detect leakage in the drywell, and (d) this instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety-related system in a post-accident situation. Therefore, we find the Category 3 instrumentation provided acceptable.

3.3.3 Radiation Level in Circulating Primary Coolant

The applicant states that the Category 3 post-accident sampling system provides a means of obtaining samples of reactor coolant and determining the status of fuel cladding. The applicant also states that the radiation monitors in the condenser off-gas and the main steamlines provide information on the status of fuel cladding when the plant is not isolated. Monitoring the primary containment radiation and containment hydrogen concentration provide this information when the plant is isolated.

Based on the alternate instrumentation provided by the licensee, we conclude that the instrumentation supplied for this variable is adequate and, therefore, acceptable.

3.3.4 Radiation Exposure Rate

Regulatory Guide 1.97, Revision 2, specifies instrumentation for this Type C variable. The applicant's position is that this variable need not be implemented. Revision 3 of Regulatory Guide 1.97 (Reference 7) states that exposure rate monitors inside buildings for detecting containment breach were deleted from the guide. Therefore, we find the applicant's position for this Type C variable acceptable.

Regulatory Guide 1.97, Revision 2, specifies instrumentation for this Type E variable. The stated range for this Category 2 instrumentation is 10^{-1} to 10^{4} R/hr. The applicant's position is that no access to a harsh environment area to service safety-related equipment following an accident is required; and that long-term accessibility will be evaluated with portable radiation survey instruments and containment atmosphere sampling and analysis. However, the applicant has provided Category 3

instrumentation for this variable with a range of 0.1 mR/hr to 10 R/hr. The applicant states that these will be used only where the anticipated radiation levels are within the instrument range.

Regulatory Guide 1.97, Revision 3, specifies Category 3 instrumentation for this variable. Therefore we find that Category 3 instrumentation is adequate for this variable. The applicant has stated that this instrumentation will be used only where they are expected to remain on scale following an accident. Based on this statement, we find the instrumentation provided for this variable acceptable.

3.3.5 Emergency Core Cooling Flow

The applicant has deviated from the recommendations of Regulatory Guide 1.97 for measuring the flow of the following systems:

a. Reactor Core Isolation Cooling (RCIC)

b. High Pressure Coolant Injection (HPCI)

c. Core Spray (CS)

d. Low Pressure Coolant Injection (LPCI).

Regulatory Guide 1.97 recommends Category 2 instrumentation for these variables, each with a range of 0 to 110 percent of design flow.

The applicant discusses, as a deviation, a potential for flow diversion for each of the four systems. This diversion could be caused by open valves in branch lines downstream of the flow measuring elements. The applicant concludes that the instrumentation for measuring the flow for these systems is adequate since it meets the intent of the regulatory guide and because the valve position is known and the valves close automatically on an accident signal. We find the instrumentation acceptable in this regard.

The flow instrumentation for the HPCI measures to 107 percent of design flow (6,000 gallons per minute). The existing range is adequate to provide the necessary accident and post-accident information. Therefore this is an acceptable deviation from Regulatory Guide 1.97.

3.3.6 Standby Liquid Control System Flow

Regulatory Guide 1.97 specifies Category 2 instrumentation with a range of 0 to 110 percent of design flow for this variable. The Hope Creek Station does not measure this variable directly. The pump discharge header pressure will indicate pump operation to the operator. Other parameters that can be monitored to verify system operation include: level decrease in the boric acid storage tank, neutron flux, pump motor contactor position (or running current), and squib valve continuity indication. The applicant maintains that these parameters are sufficient to establish that there is flow in the standby liquid control system.

The applicant uses positive displacement pumps for the standby liquid control system. High pump pressure indicates flow blockage and erratic or low pressure indicates a line break. We find that the above indications are valid for an alternate standby liquid control system flow indication.

3 3.7 Cooling Water Temperature to Engineered Safety Features (ESF) System Components

Regulatory Guide 1.97 recommends instrumentation for this variable with a range from 32 to 200°F. The applicant identifies this instrumentation for the safety auxiliary cooling system, with a range from 65 to 95°F (FSAR, Table 7.5-1, Amendment 5). No justification was given by the applicant for this deviation.

Table 9.2-3 of the FSAR lists the design water outlet temperature minimum at 65°F, the maximum at 95°F. This corresponds to the range of the instrumentation supplied. Therefore, we find the supplied range acceptable.

3.3.8 Reactor Building or Secondary Containment Radiation

Regulatory Guide 1.97 recommends that this variable be monitored with Category 2 instrumentation; the applicant's position is that secondary containment area radiation is not an appropriate parameter to use for assessing primary containment leakage or detecting significant releases.

The applicant reports that the use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations results in ambiguous indications. This is due to the radioactivity in the primary containment, the radioactivity in the fluids flowing in emergency core coolant system piping and the amount and location of fluid and electrical penetrations. The applicant concludes that the use of the plant noble gas effluent monitors is the proper way to accomplish the purpose of this variable.

The alternate instrumentation for this variable is provided by the Category 2 filtration, recirculation and ventilation system vent radiation monitoring system. It has a range of 10^{-6} to 10 μCi/cc beta and 5 to 10^5 μCi/cc gamma. This measures the radiation levels in the exhausts from the reactor building and refueling floor area in the post-accident situation. Normal monitoring of the exhausts from these areas is done by the south plant vent radiation monitoring system. This is also a Category 2 system that includes normal and extended ranges.

We find the alternate instrumentation supplied for this variable acceptable.

3.3.9 Accident Sampling (Primary Coolant Containment Air and Sump)

Regulatory Guide 1.97 recommends sampling and onsite analysis capability for the reactor coolant system, containment sump, ECCS pump room sumps and other similar auxiliary building sump liquids, the containment sump and containment air. The applicant's post-accident sampling facility provides sampling and analysis. However, there are deviations from the following recommendations.

1. The sumps are not sampled

2. Dissolved hydrogen or total gas analysis capability is not included.

The applicant takes exception to Regulatory Guide 1.97 with respect to post-accident sampling capability. This exception goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item II.B.3.

4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to or is justified in deviating from Regulatory Guide 1.97, with the following exceptions:

1. Neutron flux--the applicant's present instrumentation is acceptable on an interim basis, until Category 1 instrumentation is developed and installed (Section 3.3.1).

5. REFERENCES

1. NRC letter, D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737, Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Public Service Electric and Gas Company letter, R. L. Mittl to Director of Nuclear Reactor Regulation, NRC, "Generic Letter 82-33," April 15, 1983.

5. Hope Creek Generating Station Final Safety Analysis Report, Public Service Electric and Gas Company, Amendment 7, August 1984.

6. Public Service Electric and Gas Company Letter, R. L. Mittl to Director of Nuclear Reactor Regulation, NRC, "SER Open Item 6-Additional Information, Post-Accident Monitoring Instrumentation," May 14, 1985.

7. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 3, NRC, Office of Nuclear Regulatory Research, May 1983.

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Supplement No. 2 to the Safety Evaluation Report on the application filed by Public Service Electric and Gas Company as applicant for itself and Atlantic City Electric Company, as owners, for a license to operate Hope Creek Generating Station has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Lower Alloways Creek Township in Salem County, New Jersey. This supplement reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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