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

SEP 6 1983

Docket No.: 50-354

Mr. R. L. Mittl, General Manager
Nuclear Assurance & Regulation
Public Service Electric & Gas Company
80 Park Plaza T16D
Newark, New Jersey 07101

Dear Mr. Mittl:

Subject: Hope Creek OL Safety Review - Requests for Additional Information

Enclosures 1 through 7 to this letter identify additional information required for our review of the safety aspects of your application for an operating license for the Hope Creek Generating Station. The enclosures by subject area, are as follows:

Enclosure	Subject	SRP/FSAR
1	Equipment Qualification	3.10
2	Chemical Technology	5.4.8, 6.1.1, 9.1.2, 9.1.3, 9.3.2, 10.4.6
3	Component Integrity	5.3.1, 5.3.2, 5.3.3
4	Procedures & Systems	14.2.4, 14.2.11, 14.2.12, 14.2.13
5	Core Thermal Hydraulics	4.4
6	Inservice Inspection	5.2.4, 6.6
7	Effluents Treatment	6.5.1, 10.4.2, 10.4.3, 11.2, 11.3, 11.4, 11.5 15.7.3

Additional requests for additional information will be transmitted to you as we complete our reviews of the remaining sections.

Previous requests for additional information were transmitted to you by letters dated August 4, 10, 25, and 29, 1983. Consistent with the licensing review schedule for Hope Creek, responses to all requests for additional information should be submitted as changes to the FSAR by October 31, 1983.

If you have any questions c ncerning th enclosed requests for additional information, please call the Licensing Project Manager, Dave Wagner, at (301) 492-8525.

Sincerely,

Original signed by

A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing

Enclosures:
As stated

cc: See next page

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Hope Creek

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271.2 (SRP 3.10) Provide a brief description of the differences between the actual pump motor assembly and the "similar ECCS pump motor design assembly" tested to the requirements of IEEE-344-1975. (Page 3.9 - 45; #3.9.2.3.2.7)

271.3 (SRP 3.10) Clarification is needed regarding the level of testing in support of qualification by analysis only. For example, the RCIC pump assembly, SLC pump and motor assembly, and HPCI pump assembly are qualified by analysis only. However, in #3.9.2.4.2.2 it is indicated that dynamic testing is used to confirm operability of equipment needed during and after seismic events. Clarify if this general statement on dynamic testing includes the above equipment, and to what extent if it does. Also clarify what range of operating conditions are used for in-shop tests described in #3.9.3.2.1 (Page 3.9-45; #3.9.2.3.2.8, .10 and .15).

271.4 (SRP 3.10) The FSAR indicates that Non-NSSS active valve "operability qualification of motor, air and hydraulic operators" are to IEEE-382-1972. This standard applies only to electric actuators and the 1972 issue is not acceptable to the staff.

For those electric actuators qualified to IEEE 382-1972, describe the criteria for selecting the representative operator. Also, indicate how aging (thermal, seismic and vibrational cycles), seismic qualification (test input motion) and acceptable margin (+10% for the required response spectrum) is addressed.

For other than electric operators (air, hydraulic) indicate the bases and reference the standards used for qualification. (Page 3.9-81; #3.9.3.2.7.2).

271.5 (SRP 3.10) Clarify if the support structures for entire pump-driver assemblies are included in the shop tests and the other tests described. (Page 3.9-91; #3.9.3.4.2)

271.6 (SRP 3.10) The equipment listed in Table 3.10-3 is qualified by analysis only. Describe the test data and experience data used in support of the qualification analysis. (Pages 10-4: #3.10.2.1)

271.7 (SRP 3.10) Indicate what equipment will see hydrodynamic frequencies and indicate the frequency range to which they were qualified. (The discussions in Section 3.9 and 3.10 of the FSAR addresses test frequencies only up to 33 hertz). (Page 10-5; #3.10.2.2)

271.8 (SRP 3.10) Tables 3.10-1 through 4 and the text of #3.10 appear not to include the standby diesel generator (SDG) active starting and cooling system equipment. Provide information identifying the qualification for the air starting and cooling system equipment. (Page 3.10). (The system is shown in Table 3.2-1 and discussed in Section 9.5.6 but without any qualification information)

271.9 (SRP 3.10) Identify any manually operated valves which are required to change position for any safety system to perform its function; indicate the impact of its failure on safety function.

271.10 (SRP 3.10) Identify any safety related deep draft pumps in the plant.

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281.4 (5.4.8) (10.4.6) Verify that the initial total capacity of new demineralizer resins (condensate and primary coolant) will be measured and describe the method to be used for this measurement (Regulatory Position C.3 of Regulatory Guide 1.56, revision 1).

281.5 (10.4.6) Describe the method of determining the condition of the demineralizer units (see p. 10.4-20 of FSAR) so that the ion exchange resin can be replaced before an unacceptable level of depletion is reached (Regulatory Position C.4 of Regulatory Guide 1.56, revision 1). Describe the method by which (a) the conductivity meter readings for the condensate cleanup system will be calibrated, (b) the quantity of the principal ions likely to cause demineralizer breakthrough will be calculated, (c) the flow rates through each demineralizer will be measured, and (d) the accuracy of the calculation of resin capacity will be checked.

281.6 (5.4.8) (10.4.6) Indicate the control room alarm set points of the conductivity meters at the inlet and outlet demineralizers in the condensate and reactor water cleanup systems when either (Regulatory Position C.5 of Regulatory Guide 1.56, revision 1):

a. The conductivity indicates marginal performance of the demineralizer system; or

b. The conductivity indicates noticeable breakthrough of one or more demineralizers.

281.7 (5.4.8) (10.4.6) Indicate the reactor coolant chemistry limits and corrective action to be taken if the conductivity, pH, or chloride content, as established in the Technical Specifications, is exceeded. Describe the chemical analysis methods to be used for the determination of these values. (Regulatory Position C.6 of Regulatory Guide 1.56, revision 1).

281.8 (10.4.6) Describe the water chemistry control program to assure maintenance of condensate demineralizer influent and effluent conductivity within the limits of Table 2 of Regulatory Guide 1.56, revision 1. Include conductivity meter alarm set points and the corrective action to be taken if the limits of Table 2 are exceeded.

281.9 (10.4.6) In accordance with Regulatory Position C.1 of Regulatory Guide 1.56 revision 1, describe the sampling frequency, chemical analyses, and established limits for purified condensate dissolved and suspended solids that will be performed and the basis for these limits.

281.10 (10.4.6) Tests by EPRI have shown that intergranular stress corrosion cracking (IGSCC) can be inhibited by keeping the level of impurities in the primary coolant low and the oxygen concentration around 20 ppb. Describe how you will keep the concentration of impurities and of oxygen to a level below where IGSCC is initiated. Describe any plans being made for oxygen control by hydrogen addition.

281.11 (9.1.3) Regarding the Spent Fuel Pool Cleanup System, provide the following information:

Describe the samples and instrumentation and the frequency of the measurements that will be performed to monitor (a) the spent fuel pool water purity and (b) the need for ion exchanger resin and filter replacement. State the chemical and radiochemical limits to be used in monitoring the spent fuel pool water and for initiating corrective action. Provide the basis for establishing these limits. Your response should consider factors such as: gross gamma and iodine activity, demineralizer and/or filter differential pressure, decontamination factor, pH and crud level.

281.12 (6.1.1) Demineralized water from the condensate storage tank or the suppression pool, with no additives, is used in the containment sprays and to inject core cooling water. Indicate the limits you will place on the conductivity, the chlorides and the pH of this water to minimize stress-corrosion cracking of unstablized austenitic stainless steel components.

281.13 (9.1.2) Identify the materials, including the neutron absorbing material (poison), used in the fabrication of the high density spent fuel storage racks and all other structural components wetted by the pool water. Indicate how the poison-containing cavities are vented.

281.14 (9.1.2) Provide details of the materials monitoring program for the spent fuel pool, including type of samples used and frequency of inspection.

281.15 (9.3.2) The information provided on the Post Accident Sampling System (PASS) is inadequate to demonstrate compliance with NUREG-0737, Item II.B.3. Provide information that satisfies the criteria in the attachment.

ATTACHMENT NO. 1 TO
POST ACCIDENT SAMPLING SYSTEM
NUREG-0737, II.B.3 EVALUATION
CRITERIA GUIDELINES

The post accident sampling system will be evaluated for compliance with the criteria from NUREG-0737, II.B.3. These eleven items have been copied verbatim from NUREG-0737. The licensees submittal should include information equivalent to that which is normally provided in an FSAR. System schematics with sufficient information to verify flow paths should be included, consistent with documentation requirements in NUREG-0737, with appropriate discussion so that the reviewer can determine whether the criteria have been met. Further information pertaining to the specific clarifications of NUREG-0737, which will be considered in the reviewers evaluation are listed below. Technically justified alternatives to these criteria will be considered.

Criterion: (1) The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification: Provide information on sampling(s) and analytical laboratories locations including a discussion of relative elevations, distances and methods for sample transport. Responses to this item should also include a discussion of sample recirculation, sample handling and analytical times to demonstrate that the three-hour time limit will be met (see (6) below relative to radiation exposure). Also describe provisions for sampling during loss of off-site power (i.e. designate an alternative backup power source, not necessarily the vital (Class IE) bus, that can be energized in sufficient time to meet the three-hour sampling and analysis time limit).

Criterion: (2) The licensee shall establish an onsite radiological and chemical analysis capability to provide, within three-hour time frame established above, quantification of the following:

(a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and non-volatile isotopes);

(b) hydrogen levels in the containment atmosphere;

(c) dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

(d) Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

Clarification: 2 (a) A discussion of the counting equipment capabilities is needed, including provisions to handle samples and reduce background radiation to minimize personnel radiation exposures (ALARA). Also a procedure is required for relating radionuclide concentrations to core damage. The procedure should include:

1. Monitoring for short and long lived volatile and non volatile radionuclides such as ^{133}Xe, ^{131}I, ^{137}Cs ^{134}Cs, ^{85}Kr, ^{140}Ba, and ^{88}Kr (See Vol. II, Part 2, pp. 524-527 of Rogovin Report for further information).

2. Provisions to estimate the extent of core damage based on radionuclide concentrations and taking into consideration other physical parameters such as core temperature data and sample location.

2 (b) Show a capability to obtain a grab sample, transport and analyze for hydrogen.

2 (c) Discuss the capabilities to sample and analyze for the accident sample species listed here and in Regulatory Guide 1.97 Rev. 2.

2 (d) Provide a discussion of the reliability and maintenance information to demonstrate that the selected on-line instrument is appropriate for this application. (See (8) and (10) below relative to back-up grab sample capability and instrument range and accuracy).

Criterion: (3) Reactor coolant and containment atmosphere sampling during post accident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

Clarification: System schematics and discussions should clearly demonstrate that post accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Criterion: (4) Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification: Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is <0.1 ppm by measurement of a dissolved hydrogen residual of

$\geq$ 10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Criterion: (5) The time for a chloride analysis to be performed is dependant upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification: BWR's on sea or brackish water sites, and plants which use sea or brackfish water in essential heat exchangers (e.g. shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hours. All other plants have 96 hours to perform a chlorida analysis. Samples diluted by up to a factor of one thousand are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ____ ppm Cl (the licensee should establish this value; the number in tha blank should be no greater than 10.0 ppm Cl) in the reactor coolant system and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification no. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Criterion: (6) The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from H. R. Denton to all licensees).

Clarification: Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Criterion: (7) The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification: PWR's need to perform boron analysis. The guidelines for BWR's are to have the capability to perform boron analysis but they do not have to do so unless boron was injected.

Criterion: (8) If inline monitoring in used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samplies. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification: A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an off-site laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Criterion: (9) The licensee's radiological and chemical sample analysis capability shall include provisions to:

(a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1µ Ci/g to 10 Ci/g.

(b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification: (9) (a) Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be employed to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

(9) (b) State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Criterion: (10) Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification: The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

- Gross activity, gamma spectrum: measured to estimate core damage, these analyses should be accurate within a factor of two across the entire range.

- Boron: measure to verify shutdown margin.

In general this analysis should be accurate within $\pm 5\%$ of the measured value (i.e. at 6,000 ppm B the tolerance is $\pm$ 300 ppm while at 1,000 ppm B the tolerance is $\pm$ 50ppm). For concentrations below 1,000 ppm the tolerance band should remain at $\pm$ 50 ppm.

- Chloride: measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride the analysis should be accurate within $\pm$ 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at $\pm$ 0.05 ppm.

- Hydrogen or Total Gas: monitored to estimate core degradation and corrosion potential of the coolant.

An accuracy of $\pm$ 10% is desirable between 50 and 2000 cc/kg but $\pm$ 20% can be acceptable. For concentration below 50 cc/kg the tolerance remains at $\pm$ 5.0 cc/kg.

- Oxygen: monitored to assess coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm oxygen the analysis should be accurate within $\pm$ 10% of the measured value. At concentrations below 0.5 ppm the tolerance band remains at $\pm$ 0.05 ppm.

- pH: measured to assess coolant corrosion potential.

Between a pH of 5 to 9, the reading should be accurate within ±0.3 pH units. For all other ranges ± 0.5 pH units is acceptable.

To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX
FOR
UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituient	Nominal Concentration (ppm)	Added as (chemical salt)
I^-	40	Potassium Iodide
Cs+	250	Cesium Nitrate
Ba+2	10	Barium Nitrate
La+3	5	Lanthanum Chloride
Ce+4	5	Ammonium Cerium Nitrate
Cl^-	10	
B	2000	Boric Acid
Li+	2	Lithium Hydroxide
NO_3^-	150	
NH_4^+	5	
K+	20	
Gamma Radiation (Induced Field)	10^4 Rad/gm of Reactor Coolant	Adsorbed Dose

NOTES:

1) Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2) For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3) For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4) In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every six months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Criterion: (11) In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

(a) Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

(b) The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

Clarification: (11)(a) A description of the provisions which address each of the items in clarification 11.a should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident (i.e. sampling from a hot or cold leg loop which may have a steam or gas pocket) describe the backup sampling capabilities or address the maximum time that this condition can exist.

BWR's should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

(11)(b) A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

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251.1 (SRP 5.3.1 5.3.2 5.3.3) Appendices G and H, 10 CFR Part 50 were revised in the Federal Register on May 27, 1983 effective July 26, 1983.

a. Identify ferritic reactor coolant pressure boundary materials that do not comply with the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50.

b. For materials which cannot meet the fracture toughness requirements of Section 50.55a and Appendices G and H of 10 CFR Part 50, provide alternative fracture toughness data and analyses to demonstrate their equivalence to the requirements of 10 CFR Part 50.

c. To demonstrate conformance to Appendices G and H, 10 CFR Part 50:

(1) Provide pressure temperature limit curves for hydrostatic pressure and leak tests, heat-up, cooldown and core operations.

(2) Identify the withdrawal schedule, lead factor, test samples and materials in the Reactor Vessel Materials Surveillance Program.

(3) Indicate the reference temperature, RT_{NDT}, for materials in the reactor vessel closure flange region and the beltline regions.

(4) Indicate the chemical composition (copper, nickle and phosphorus), unirradiated upper-shelf energy, and projected end-of-life RT_{NDT} and upper-shelf energy for all beltline materials. RT_{NDT} projections are to be estimated using the "Guthrie Formula" in Commission Report SECY-82-465. Upper-shelf energy projects are to be estimated using Regulatory Guide 1.99, Rev. 1. These projects are to be for the end-of-life neutron fluence at the 1/4T and ID reactor vessel locations.

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640.2 (14.2.4) (14.2.12) Regulatory Guide 1.68 (Initial Test Program for Water-Cooled Nuclear Power Plants), Appendix A, Section 5 states the approximate power levels for the conduct of various startup tests.

1. Modify FSAR Subsection 14.2.4.5 to include a startup test plateau of approximately 50% power or provide technical justification for why this power level is not included.

2. Modify the individual startup test abstracts, or provide an appropriate table, to specify the power-to-flow test conditions at which the testing will be accomplished.

640.3 (14.2.11) Modify FSAR Subsection 14.2.11 to conform to Regulatory Guide 1.68 (Appendix B) such that copies of all preoperational test procedures will be available for examination by the NRC regional personnel approximately 60 days prior to the scheduled performance of the tests, and not less than 60 days prior to the scheduled fuel loading date, copies of procedures for fuel loading, initial startup tests, and supporting activities will be available. Drafts of these procedures should be made available as early as practical. (Examination by NRC personnel does not constitute approval of the procedures. The possession of such procedures by NRC personnel should not impede the revision, review, and refinement of the procedures by the applicant.)

640.4 (14.2.12) Modify your FSAR Chapter 14 submittal to include an index of preoperational and startup tests as stated in FSAR Subsection 14.2.12.1.

640.5 (14.2.12) Modify the acceptance criteria provided in the individual test descriptions for each preoperational and startup test listed in FSAR Subsection 14.2.12 such that for all tests subject to Quality Assurance Program requirements (FSAR Chapter 17 - which includes those structures, systems, and components that meet the criteria of Regulatory Guide 1.68, Positions C.1.a - C.1.f), specific acceptance criteria or a discussion of the source for the acceptance criteria to be used when test procedures are prepared is included. Each test description should provide "traceability" to acceptance criteria sources such as: specific FSAR subsections, Technical Specifications, topical reports, vendor-furnished test specifications, and/or accident analysis assumptions.

640.6 (14.2.12) Modify FSAR Subsection 14.2.12.1.2 (AE-Feedwater) to include tests of primary condensate and secondary condensate pumps in accordance with Regulatory Guide 1.68.1 (Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants), Position C.1.a, and to include major simulated transients in accordance with Position C.1.f of this guide.

640.7 (14.2.12) Expand FSAR Subsection 14.2.12.1.5 (BC-Residual Heat Removal) to include verification that the paths for the air-flow test of containment spray nozzles overlap the water-flow test paths of the pumps in order to demonstrate that there is no blockage in the flow path (Regulatory Guide 1.68, Appendix A.1.h(3)).

640.8 (1.8) (14.2.12) Modify your FSAR submittal to address the following items regarding conformance with Regulatory Guide 1.68.3 (Preoperational Testing of Instrument and Control Air Systems):

1. Delete the exception to Position C.4 of this guide in FSAR Subsections 1.8.1.68.3 and 14.2.13.4. Position C.4 applies to the system as a whole, not to each branch line.

2. FSAR Subsection 14.2.12.1.27 (KB-Instrument and Compressed Air) does not demonstrate conformance with Positions C.9, C.10, and C.11 of this guide. Either describe appropriate tests or provide technical justification for any exceptions to these positions.

640.9 (14.2.12) Modify FSAR Subsection 14.2.12.1.29 (KC-Fire Protection - Deluge) to provide assurance that:

1. Upon automatic sprinkler actuation, adequate drainage in the affected spaces is provided to preclude flooding (including expected hand-held hose volume).

2. A walk-down of plant equipment is conducted to identify potential incidences where the actuation of fire suppression systems could cause damage to or inoperability of systems important to safety.

See IE Information Notice 83-41: Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment, June 22, 1983.

640.10
(1.8)
(14.2.12)
(14.2.13)

Modify your FSAR submittal to address the following concerns regarding emergency diesel generator testing:

1. FSAR Subsections 1.8.1.108 and 14.2.13.5 state that Regulatory Guide 1.108 (Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants) is not applicable to Hope Creek. It is the staff's position that this guide is applicable to your facility. Therefore, either delete or provide justification for this statement.

2. FSAR Subsections 1.8.1.108 and 14.2.13.5 take exception to Position C.2.a(5) of Regulatory Guide 1.108. These subsections state that testing of the sequencing controls after the 24 hour test run does not subject the controls to more severe conditions than testing accomplished under other circumstances. Provide technical justification for your position or perform this test in accordance with this guide. Additionally, modify FSAR Subsection 14.2.12.1.30 (KJ-Emergency Diesel Generators) to perform a restart simulating loss of ac directly after the 24-hour run in accordance with your statement in the aforementioned FSAR subsections.

3. Modify FSAR Subsections 14.2.12.1.30 (KJ-Emergency Diesel Generators), 14.2.12.3.30 (Loss of Turbine-Generator and Offsite Power), or other test abstracts as appropriate, to:

   a. Perform the simultaneous, redundant diesel starts specified in Position C.2.b of Regulatory Guide 1.108.

   b. Include prerequisite testing to ensure the satisfactory operability of all check valves in the flow path of cooling water for the diesel generators from the intake to the discharge (see I&E Bulletin No. 83-03: Check Valve Failures in Raw Water Cooling Systems of Diesel Generators).

   c. Provide assurance that any time delays in the diesel generator's restart circuitry will not cause the supply of compressed air used to initially rotate the engine to be consumed in the presence of a safety injection signal (see I&E Information Notice Number 83-17, March 31, 1983).

640.11 (14.2.12) In accordance with Regulatory Guide 1.41 (Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments), Positions C.2 and C.3:

1. Modify FSAR Subsection 14.2.12.1.32 (PB-4160-V Class 1E Station Power), or other test abstracts as appropriate, to demonstrate the proper operation of transformer cooling under design load or describe how data from testing under available load will be extrapolated to verify cooling capability under design loading.

2. Modify FSAR Subsections 14.2.12.1.35 (PJ-250-V d-c Class 1E Power) and 14.2.12.1.36 (PB-125-V d-c Class 1E Power) to incorporate testing to verify that required Class 1E loads can be started and operated at the minimum and maximum design battery voltages. The battery chargers should not be put in use until after the 1E loads have started (IEEE 308-1980). For more information on problems with maximum battery voltage conditions, see I&E Information Notice 83-08, March 9, 1983.

3. Modify abstracts of preoperational tests involving sources of power to vital a-c buses to ensure that full-load testing, or extrapolation to full-load testing conditions, is accomplished.

4. Modify abstracts of all preoperational tests associated with d-c and on-site a-c buses to ensure that during such testing, the d-c, on-site a-c, and related loads not under test will be monitored to verify the absence of voltage.

640.12 (14.2.12) Modify FSAR Subsection 14.2.12.1.38 (QF-In-Plant Communication) to provide a description of the testing to be performed to meet the requirements of 10 CFR 50 Appendix E.IV.E, I&E Bulletin No. 80-15, and Generic Letter 82-33, or provide additional test abstracts or cross-references to describe such tests.

640.13 (14.2.12) Modify FSAR Subsection 14.2.12.3.5 (Control Rod Drive System) in accordance with Regulatory Guide 1.68, Appendix A, Paragraph 2.b to:

1. Specify the reactor pressure and flow conditions which will be used during testing, and identify the testing that will be accomplished at each test condition.

2. Specify rod withdrawal, insert, and scram time acceptance criteria.

640.14 (14.2.12) For compliance with Regulatory Guide 1.68, Appendix A.3, modify FSAR Subsection 14.2.12.3.6 (Source Range Monitor Performance and Control Rod Sequence) to ensure:

1. A neutron count rate of at least 1/2 count per second registers on the startup channels before startup begins.

2. The signal-to-noise ratio is greater than two.

3. Initial criticality will be approached on a startup rate of less than 1 decade/minute.

640.15 (14.2.12) NUREG-0694, "TMI Related Requirements for New Operating Licenses" Item I.G.1, requires applicants to perform "a special low power testing program approved by NRC to be conducted at power levels of greater than 5% for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training." To comply with this requirement, provide test descriptions and a commitment to the recommendations of the BWR Owners Group response to NUREG-0737, Item I.G.1 (Letter from D. B. Waters to D. G. Eisenhut dated February 4, 1981).

640.16 (14.2.12) Modify FSAR Subsection 14.2.12.3.12 (Reactor Core Isolation Cooling System) to include verification that the FSAR-defined RCIC steam flow setpoint is consistent with the actual startup data. For more information see I&E Information Notice Number 82-16: HPCI/RCIC High Steam Flow Setpoints, dated May 28, 1982.

640.17 (14.2.12) Modify FSAR Subsection 14.2.12.3.24 (Relief Valves) to describe or reference any confirmatory in-plant tests of safety-relief valves to be performed in compliance with NUREG-0763 "Guidelines for Confirmatory Inplant Tests of Safety-Relief Valve Discharges for BWR Plants."

640.18 (14.2.12) Modify FSAR Subsection 14.2.12.3.26 (Shutdown From Outside the Main Control Room) or other appropriate tests to demonstrate conformance with Regulatory Guide 1.68.2 (Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants), Positions 3.b and 4.a - 4.d.

640.19 (14.2.12) To meet the objectives of Regulatory Guide 1.68, Appendix A.5.j.j, modify FSAR Subsection 14.2.12.3.30 (Loss of Turbine-Generator and Offsite Power) to ensure that the loss of power is maintained long enough for plant conditions to stabilize (>30 minutes).

640.20
(14.2.12)

Our review of your test program description disclosed that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Revision 2), Appendix A, may not be adequately demonstrated by your initial test program. Expand FSAR Subsection 14.2.12 to address the following items:

NOTE: Inclusion of a test description in FSAR Chapter 14 does not necessarily imply that the test becomes subject to FSAR Chapter 17 Quality Assurance Program controls. Certain tests, performed prior to fuel loading to verify system operability, may be referred to as "acceptance tests" to distinguish them from "preoperational tests" subject to FSAR Chapter 17 test control.

PREOPERATIONAL TESTS

R.G. 1.68 APPENDIX A	FSAR SUBSECTION	DESCRIPTION
1.d (1)	10.4.4	Turbine Bypass Valves
1.d (7)	10.3.2	Branch Steam Isolation Valves, Nonreturn Valves
1.d (9)	9.2.6	Condensate Storage System
1.e (5)	10.2.2	Steam Extraction System
1.e (6)	10.2.2	Turbine Stop, Control, and Intercept Valves
1.e (7)	10.4.1.5.1	Condenser Hotwell Level Control System
1.e (8)	10.4.7.2.1	Condensate System
1.e (10)	10.4.2	Feedwater Heaters
	10.4.7	Feedwater Drain System
1.e (12)	10.4.2	Main Condenser Evacuation System
1.f (1)	10.4.5	Circulating Water System
1.f (2)	10.4.5	Cooling Tower
1.g (2)	9.5.3	Lighting System

1.h	5.4.4	Main Steam Line Flow Restrictors
1.h (8)	6.3.2.2.6	ECCS Discharge Line Fill Network
1.i (1)	3.8.2 6.A.1	Containment Design Overpressurization and Vacuum Tests
1.i (2)	6.2.4	Containment Isolation System
1.i (8)	7.6.1.8	Containment Isolation Logic
1.i (16)	9.4.5	Primary Containment Ventilation System
1.i (19)	6.2.6.5.1	Drywell to Pressure Suppression Chamber Atmosphere Bypass Area Test
1.j (5)	7.6.1.3	Leak Detection System
1.j (6)	7.7.1.3	Loose Parts Monitoring System
1.j (8)	10.2.3	Electrohydraulic Control System
1.j (14)	9.2.6	Condensate Transfer System
1.j (19)	7.4.1.4	Remote Shutdown Panel
1.j (21)	7.7.1.4	Reactor Mode Switch and Associated Functions
1.j (25)	7.7.1.5	Process Computer System
1.1 (1)	11.2	Liquid Radwaste
1.1 (2)	11.3	Gaseous Radwaste
1.1 (3)	11.4	Solid Radwaste
1.1 (5)	11.5.2.2.6	Isolation of Condenser Offgas
1.1 (6)	11.5.2.2	Isolation of Ventilation Systems
1.1 (7)	11.5.2.2.5	Isolation of Liquide Radwaste
1.1 (8)	9.3.2 11.5	Plant Sampling Systems
1.m (2)	9.1.4	Fuel Handling System

1.m (3)	9.1.2	Operability and leak tests of sectionalizing devices and drains and leak tests of gallows or bellows in the refueling canal and fuel storage pool.
1.m (4)	9.2.2	Static (125%) and dynamic (100%) load tests of cranes, hoists, and associated fuel storage and handling systems.
1.m (5)	9.1.4	Fuel Transfer Devices
1.n (2)	9.2.7	Turbine Building Chilled Water System
1.n (4)	9.2.3	Makeup Demineralization System
1.n (5)	9.3.4 11.5	Reactor Coolant Sampling
1.n (8)	10.4.3	Steam Seal System
1.n (9)	9.3.3	Equipment and Floor Drain System
1.n (10)	10.4.6	Condensate Cleanup System
1.n (14) (e)	9.4.2	Reactor Building HVAC
1.n (18)		Heat Trace and Freeze Protection Systems
1.o (1)	9.1.5.2.1	Polar crane static (125%) and dynamic (100%) load tests
STARTUP TESTS		
2.c	7.2.1.1.4	Reactor Protection System
2.d	5.2.5	Leak Detection System
	9.3.3	Equipment and Floor Drainage System
5.n	7.7.1.7	Loose Parts Monitoring
5.s	10.2.3	Electrohydraulic Control System
	10.4.1.5.1	Hotwell Level Control System
5.o	5.2.5	Leak Detection System

5.w	9.4.5	Penetration Coolers. For those penetrations where coolers are not used, include a test description for a containment penetration concrete temperature survey to assure that penetrations will not subject concrete to temperatures over 150°F, in accordance with FSAR Subsection 9.4.5.1.c.
5.g.g	15.8	ATWS Test
5.i.i	15.3.2	Reactor Coolant Pump Trip/Reactor Coolant Flow Control Valve Closure
5.k.k	15.1.1	Loss of, or Bypass of, Feed Heaters

640.21 (1.8) (14.2.12) (14.2.13) Certain exceptions to Regulatory Guide 1.68 contained, in FSAR Subsections 1.8.1.68 and 14.2.13.1, need clarification or are unacceptable:

1. The exception to Position C.9 should be deleted. Position C.9 does not specify that preoperational testing be included in the Summary Startup Report.

2. The clarification to Appendix A, Paragraphs 1.a (1), 1.e, and 1.h (1), states that certain tests not performed prior to fuel load will be performed after fuel load. For portions of any preoperational tests (including review and approval of test results) which are now intended to be conducted after fuel loading: (a) list each test; (b) state what portions of each test will be delayed until after fuel loading; (c) provide technical justification for delaying these portions; and (d) state when each test will be completed. Note that this exception references different paragraphs of this guide in the two subsections.

3. The exception to Appendix A, Paragraphs 1.a (3), 4.s, and 5.p, states that no reactor internal vibration tests are planned following fuel load. FSAR Subsection 1.8.1.20 and 14.2.13.2 state conformance with Regulatory Guide 1.20 (Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing). Position C.2.2.2.e of this guide states that the fuel assemblies should be in position during the vibration tests. Either modify your test program accordingly or provide technical justification for not performing these tests with fuel in place.

4. The exception to Appendix A, Paragraph 1.g (2), states that emergency sources will only be tested at normal voltage. The regulatory position states that emergency loads, not sources, should be tested at maximum and minimum design voltages. Reference test abstracts where this testing is accomplished or provide technical justification for not performing tests over the design range of operating voltages. If emergency loads are to be started and operated only at normal voltage during preoperational testing, confirm that measurements will be taken at each terminal load connection and analyzed in accordance with Branch Technical Position PSB-1. Ensure that this analysis includes extrapolation to design source voltages (minimum and maximum). The intent is to provide assurance that the voltage supplied at the terminal load connection is not so much lower than the source voltage that the equipment may end up operating outside its design voltage range, even though source voltage is within its design range.

5. The exception to Appendix A, Paragraph 1.h (10), states that there is no practical way to verify the maximum heat removal capability of the ultimate heat sink (UHS). Provide testing or reference analyses which demonstrate that the UHS meets the performance criteria of Regulatory Guide 1.27 (Ultimate Heat Sink For Nuclear Power Plants). Also, provide test descriptions or reference analyses that ensure the station service water pumps used for long-term post-accident cooling will have adequate net positive suction head, and that there will be an absence of vortexing, at the minimum postulated river level.

6. The exception to Appendix A, Paragraph 4.m, states that there is no startup test planned for the MSIV leakage control system. Provide appropriate preoperational or startup test abstracts to demonstrate the operability of the MSIV leakage control system at rated temperature and pressure or provide technical justification for not performing this testing.

7. The exception to Appendix A, Paragraph 5.q, states that no startup tests of the failed fuel detection systems are planned. Provide appropriate startup test abstracts to verify proper operation of the failed fuel detectors system at 25% and 100% power, or provide documentation that the surveillance test properly establishes a baseline at these levels.

640.22 (14.2) Modify Figure 14.2-3 to identify the length and time frame allocated for the preoperational test phase (Phase II). (Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports For Nuclear Power Plants, Paragraph 14.2.11).

640.23 (14.2.12) To help facilitate approval of future changes to the Hope Creek Initial Test Program, list and provide technical justification for any startup tests or portions of startup tests which you believe should be exempted from the license condition requiring prior NRC notification of major test changes to tests described in FSAR Chapter 14. Such a list should include those tests not necessary to verify the proper design, construction, or performance of systems, structures, or components important to safety (fulfill General Design Criteria (GDC) functions and/or are subject to 10 CFR 50 Appendix B Quality Assurance requirements).

HOPE CREEK GENERATING STATION

492.2 (4.4) Section 4.4 of the Standard Review Plan states that the crud effects should be accounted for in the thermal-hydraulic design by including it in the CHF calculations in the core or in the pressure drop throughout th RCS. Process monitoring provisions should be capable of detecting a 3 percent pressure drop in the RC flow. The staff found that section 4.4 of the Hope Creek FSAR has not discussed crud effects and the process monitoring provision. Please provide a submittal addressing the crud effect. The assumptions used for amount of crud in design calculations and the sensitivity of operating limit MCPR and core pressure drop to variations in the amount of crud present should be addressed. However, provisions to detect crud build-up in the core are of no concern during power operation and need not be described because of the power-flow map characteristics showing a decrease in power as RC flow decreases.

492.3 (4.4) The staff is performing a generic study of the thermal-hydraulic stability characteristics of BWRs under normal operation, anticipated transients and accident conditions. The results of this study will be applied to the staff review for acceptance of stability analyses. In the interim, the staff concludes that past operating experiences and inherent thermal-hydraulic characteristics of BWRs provide a basis for accepting the stability evaluation for normal operation stability limits, natural circulation operation will be prohibited by the Technical Specifications. Any action resulting from the staff generic study will be applied to Hope Creek.

492.4 (4.4) No analysis has been presented for the operating limit MCPR or thermal-hydraulic stability characteristics for one loop operation. One loop operation will not be permitted until supporting analyses are provided and approved by the staff.

492.5 (4.4) No analysis has been presented for the core thermal-hydraulic stability. PSE&G is required to provide the calculated decay ratio as a function of power level. The submittal also should indicate the applicability of your calculated results to the specific fuel types and the cycle numbers of the Hope Creek core.

492.6 (4.4) The FSAR includes a very limited description of the loose parts monitoring system (LPMS). The staff has reviewed the FSAR regarding the LPMS program and found partial conformance with Regulatory Guide 1.333. Table 4.4.0 attached, "Summary of Review of Hope Creek LPMS," lists staff's findings on the area conforming to the Guide and areas where additional information (these with symbol I or NI) is required. The LPMS should be installed to meet the operability requirements of Regulatory Guide 1.333. The LPMS must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing. The staff requires that PSE&G commit to evaluate the system to address conformance with Regulatory Guide 1.333. The conformance evaluation should emphasize the following areas:

1. A description and evaluation of diagnostic procedures used to confirm the presence of a loose part.

2. A description of how the operators will be trained in the purpose and implementation of the system.

3. A description of system calibration, including signature analysis, evaluation of background noise and alarm settings.

4. A description and evaluation of alert level establishment procedure with consideration of background noises.

The staff also requires the commitment to provide the additional information identified in Table 4.4.0 prior to power operation. The staff will review the additional information when it becomes available. Any action resulting from the staff's review will be applied to Hope Creek at that time.

TABLE 4.4.0

SUMMARY OF REVIEW OF HOPE CREEK

RG 1.133 SECTION				HOPE CREEK LPMS
C.1	System Characteristics			
	a.	Two sensors at each natural collection region.		C
	b.	Sensitivity of 0.5 ft-lb within 3ft of sensor.		NI
	c.	Physical separation of instrumentation channels.		I
	d.	Automatic data acquisition (tape recorder).		I
	e.	Automatic comparison of signal to an alert level.		NI
	f.	Periodic system operational verification and calibration.		NI
	g.	Ability to function after seismic event.		NI
	h.	Quality of system components.		NI
	i.	Ease of repair to minimize radiation exposure.		NI
C.2.	Establishing the Alert Level			
	a.	Logic to recognize LP in presence of noise.		NI
	b.	Override of noise caused by control rod movement, etc.		NI
	c.	Alert level a function of plant operating conditions.		NI
	d.	Compensation for different background noise on sensors.		NI
C.3.	Using the Data Acquisition Modes			
	a.	Manual Mode		
		(1)	Pre-op tests to establish alert level.	NI
		(2)	Startup and power operation.	
			a. Submit alert level within 90 days after startup.	NI
			b. Perform channel check each 24 hours.	NI
			c. Listen to audio output each 7 days.	NI
			d. Perform functional test each 31 days.	NI
			e. Verify background noise each 92 days.	NI

RG 1.133 SECTION	HOPE CREEK LPMS
(3) Verify channel calibration each 18 months.	NI
b. Automatic data recording when alert level is exceeded	I
C.4. Content of Safety Analysis Reports	
a. Sensor type, location, mounting, and criteria for these.	I
b. Description of data acquisition, recording, and calibration.	I
c. Major sources of extraneous noise.	NI
d. Quality assurance of data.	NI
e. Description of alert level determination and alert logic	NI
f. Reference to Technical Specification.	NI
g. Description of diagnostic procedures used to confirm loose part.	I
h. Channel check procedures.	NI
i. Maintenance procedures to minimize radiation exposure.	NI
j. Training program.	NI
k. Verification that LPMS will function after a seismic event.	NI
C.5. Technical Specification for Loose-Part Detection System.	NI
C.6. Notification of a Loose Part.	NI

KEY:
- C - Conformance with RE 1.133.
- NC - Nonconformance with RG 1.133.
- I - Insufficient information provided.
- NI - No information provided.
- NA - Not applicable at this time.

Hope Creek Generating Station

250.1 (SRP 6.6 & 5.2.4) To provide our input to SER Sections 5.2.4 and 6.6, the staff requires that the PSI Program Plan be submitted for review. Provide a schedule defining when the entire PSI Program will be submitted. The PSI Program should include reference to the ASME Code Section XI Edition and Addenda that will be used for the selection of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable code, and the examination isometric drawings.

The Hope Creek FSAR, Section 6.6.1, states in part that "...the extent of examination (selection of welds to be examined) for Class 2 piping in the ECCS and RHR systems has been determined by the requirements of Subarticle IWC-1220, Table IWC-2520 Categories C-F and C-G and Paragraph IWC-2411 of ASME XI, 74S75." Paragraph 10 CFR 50.55a(b)(2)(iv) requires that piping welds in the Residual Heat Removal (RHR) Systems, Emergency Core Cooling (ECCS) Systems, and Containment Heat Removal (CHR) Systems be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in IWC-1220. The control of water chemistry to minimize stress corrosion described in Paragraph IWC-1220(c) of Section XI is not an acceptable basis for exempting ECCS, RHR, and CHR components from examination because practical evaluation, review, and acceptance standards cannot be defined. To satisfy the inspection requirements of General Design Criteria 36, 39, 42, and 45, the preservice inspection program must include volumetric examination of a representative sample of welds in the RHR, ECCS, and CHR Systems.

Plans for preservice examination of the reactor pressure vessel welds should address the degree of compliance with Regulatory Guide 1.150. Discuss the method to be used to qualify the examination procedures to assure finding service-induced flaws on the inside surface of the vessel.

The Hope Creek FSAR Sections 5.2.4, 5.3.3.7, and 6.6 describing the inservice inspection program are not consistent with 10 CFR 50.55a(g) which requires that the inservice examinations comply with the requirements in the latest edition and addenda of the ASME Code incorporated by 10 CFR 50.55a(b) on the date 12 months prior to the date of issuance of the operating license and, therefore, should be revised.

250.2 (SRP 6.6 & 5.2.4) Describe the measures taken to ensure that austenitic stainless steel piping welds, which have been determined to be "service sensitive" to IGSCC as defined in NUREG-0313, are examined using effective techniques and the methods of assuring adequate examination sensitivity over the required examination volume. Discuss the preservice examination criteria used to record, report, and plot geometric or metallurgical ultrasonic indications in "service sensitive" piping systems to assure correlations of baseline data with inservice inspection results.

The ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978 and 1980 Edition specifies the use of Appendix III of Section XI for ferritic piping welds. If this requirement is not applicable (for example, for austenitic piping welds), ultrasonic examination is required by Section XI to be conducted in accordance with the applicable requirements of Article 5 of Section V, as amended by IWA-2232. A technical justification is required if any alternatives are used. If Section XI, Appendix III, Supplement 7, will be used for the examination of austenitic piping welds, discuss the following:

1. All modifications permitted by Supplement 7.

2. Methods of qualifying the procedure for examination through the weld (if complete examination is to be considered for examination conducted with only one side access).

When using either Article 5 of Section V or Appendix III of Section XI for examination of either ferritic or austenitic piping welds, the following should be incorporated:

1. Any crack-like indication, regardless of ultrasonic amplitude, discovered during examination of piping welds or adjacent base metal materials should be recorded and investigated by a Level II or Level III examiner to the extent necessary to determine the shape, identity, and location of the reflector.

2. The Owner should evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometrical or metallurgical in nature.

250.3 (SRP 6.6 & 5.2.4) All preservice examination requirements defined in Section XI of the ASME Code that have been determined to be impractical must be identified and a supporting technical justification for requests for relief must be provided. Indicate an anticipated date for submittal of these relief requests. The relief requests submittal should include at least the following information:

1. For ASME Code Class 1 and 2 components, provide a table similar to IWB-2500 and IWC-2500 confirming that either the entire Section XI preservice examination was performed on the component or relief is requested.

2. Where relief is requested for pressure-retaining welds in the reactor vessel, identify the specific welds that did not receive a 100% preservice ultrasonic examination and indicate the extent of the examination that was performed.

3. Where relief is requested for piping system welds (Examination Category B-J, C-F, and C-G), provide a list of the specific welds that did not receive a complete Section XI preservice examination including drawing or isometric identification number, system, weld number, and physical configuration (e.g., pipe-to-nozzle welds, etc.). Indicate the extent of the preservice examination that was performed. When the volumetric examination was performed from one side of the weld, discuss whether the entire weld volume and the heat affected zone (HAZ) and base metal on the far side of the weld were examined. State the primary reason that a specific examination is impractical (e.g., support of component restricts access, fitting prevents adequate ultrasonic coupling on one side, component-to-component welds prevents ultrasonic examination, etc.). Indicate any alternative or supplemental examinations performed and methods of fabrication examination.

250.4 (SRP 6.6) FSAR Section 6.6.8 states that since all high energy fluid system piping penetrating containment is NRC Quality Group A, and ASME Code Section III, Class 1, an augmented inservice inspection is not applicable to Hope Creek. Augmented inservice inspection should be performed on welds of high energy piping in the containment penetration region where pipe breaks are not postulated and where the effects of breaks can not be accommodated as described in FSAR Section 3.6.

Hope Creek Generating Station

460.11 (Table 3.2-1) Add the following entries to your list contained in Table 3.2-1, "HCGS Design Criteria Summary" of the Hope Creek Generating Station (HCGS) Final Safety Analysis Report (FSAR):

a. All Effluent Monitors

b. Control Room Emergency Filter System (CREFS)

460.12 (SRP 6.5.1) With regard to the ESF filter systems, i.e., the CREFS and the reactor building Filtration, Recirculation and Ventilation System (FRVS), provide information on the following items:

a. Compliance with the minimum instrumentation, readout, recording and alarm provisions for these systems as required by Table 6.5.1-1 of SRP 6.5.1, "ESF Atmosphere Cleanup Systems," Rev. 2, July 1981. For each requirement identified in the table for which an exception is taken, justify the exception.

b. Clarify whether the CREFS will be automatically activated by a redundant ESF signal as required by Position C.2.i of Regulatory Guide 1.52, Rev. 2, March 1978, "Design, Testing and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of

Light-Water-Cooled Nuclear Power Plants." If not, provide justification for the exception taken.

c. The test procedures and criteria for the ESF filters that you have currently described in the HCGS FSAR are incomplete (for example, you have not described the laboratory testing for these filters). Complete these procedures and the test criteria. In this context, you may note that at the time when the Technical Specifications (TS) for the operation of HCGS are issued, the TS relating to these ESF filters will be the same as the generic TS for such filters in a BWR.

460.13 (SRPs 10.4.2 & 10.4.3

With regard to the Main Condenser Evacuation System (MCES) and the Turbine Gland Sealing System (TGSS):

a. Clarify whether the components will be designed to Quality Group D as defined in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-Steam and Radioactive Waste Containing Components of Nuclear Power Plants."

b. Clarify whether the components will conform with Regulatory Guides 1.33 and 1.123, "Quality Assurance Program Requirements" and "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants," respectively.

For items a and b, provide justifications for any exception that you take (the exceptions that you have currently taken regarding Regulatory Guide 1.123 are not satisfactory).

460.14 (SRPs 11.2 11.3 & 11.4) The information you have currently provided relating to compliance with Regulatory Guide 1.143, Rev. 1, October 1979,"Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants is incomplete (for example, you have not explicitly referred to this guide in your description of the Solid Waste Management under Section 11.4 of the HCGS FSAR; also you have not listed the inside and outside tanks that have provisions to monitor liquid levels). Provide a table under Section 1.8 comparing the design features of the liquid, gaseous and solid radwaste systems with each regulatory position of the above guide. For each position identified in the guide for which an exception is taken, justify the exception. If information has already been provided for the individual regulatory positions, cross references to the applicable sections of the FSAR will be adequate.

460.15 (SRPs 11.2 & 11.3) Clarify whether the cost-benefit analysis you have provided on June 1, 1976 for your plant is valid now. If it is, refer to that cost-benefit analysis in Sections 11.2 and

11.3 of the HCGS FSAR. If, however, the assumptions and methodology that you used earlier have changed now, provide updated cost-benefit analyses for the liquid and gaseous radwaste management systems in the HCGS FSAR to demonstrate compliance with the applicable position of Appendix I to 10 CFR Part 50.

460.16 (SRP 11.4) With regard to the Solid Waste Management System (SWMS), provide the following:

a. Describe the provisions you have for complying with the Branch Technical Position ETSB 11-3, Rev. 2, July 1981, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Plants." Your description should include 1) curbs and drainage provisions for containing radioactive spills, 2) heat tracing for evaporator concentrate piping and tanks that are likely to solidify at ambient temperatures, 3) flushing connections wherever appropriate, 4) provisions for controlling release of airborne dusts generated during compaction process for the "dry" solid wastes, and 5) appropriate waste storage capacities for tanks accumulating spent resins from Reactor Water Cleanup System and other sources and filter sludges per the above mentioned BTP (position III.1).

b. Since the volume of the space you have indicated for temporary storage of "wet" solidified wastes in the auxiliary building (Section 11.4.1.2 of the HCGS FSAR) falls short of the required one month minimum storage capacity for such wasts (expected to be $\sim$ 40,000 CF/yr for HCGS per staff's projections), explain what contingency provisions you have for the storage of solidified "wet wastes" should they exceed your expectation.

c. Clarify how you propose to implement the 10 CFR Part 61 requirements relating to licensed burial facility's acceptance criteria.

470.17 (SRP 11.5) With regard to process and effluent radiological monitoring and sampling systems:

a. Provide in tabular columns sampling frequency, minimum analysis frequency and sensitivity in uCi/cc for the following airborne effluent and process streams:

   1. grab sampling of the effluents for principal gamma emitters and tritium for the north and south plant vents, Reactor Building Ventilation System Exhaust (RBVSE) when the drywell is purged, and the FRVS Vent exhaust during periodic operational checks;

2. grab sampling of the effluent for principal gamma emitters for the off-gas system;
3. grab sampling for iodine, particulates and alpha activity in process streams for radwaste building vent, turbine building vent, fuel handling area vent, evaporator vents, and other applicable vent gas systems such as radwaste area of the auxiliary building, etc.; and
4. continuous sampling of the effluents for iodine, particulates and gross alpha for north and south plant vents and the FRVS when the FRVS undergoes periodic operational checks.

In this context, you may note that at the time when the TS for HCGS are formalized, the TS will require you to release the drywell purge exhaust only after the drywell purge exhaust concentrations are reduced to within the 10 CFR Part 20 limits and, in addition, stay within these limits during the entire purge operation and/or the purge exhaust always goes through an exhaust treatment system which complies with the regulatory positions given in Regulatory Guide 1.52, Rev. 2, March 1978.

b. For liquid effluents and process streams, provide information on sampling frequency, minimum analysis frequency, type of activity analysis and lower limits of detection in tabular columns. Your information should include sampling and analysis provisions for effluents from the Station Service Water System (SSWS) and all other liquid effluents; it should also include grab sampling provisions in the process liquid streams for the Safety Auxiliary Cooling System (SACS), Turbine Auxiliary Cooling System (TACS), laboratory and sample system waste systems and spent and refueling pool treatment systems.

c. Clarify whether the design criteria for the radiological effluent monitors will conform with the manufacturer's standard per ANSI N13.10-1974 and quality assurance criteria set forth in Sections 4 and 6 of the Regulatory Guide 1.143, Rev. 1, October 1979. If they do not, provide justifications for the deviations.

d. Clarify whether <u>all</u> the airborne effluent pathways have effluent system flow monitors and sampler flow rate measuring devices; also clarify whether <u>all</u> the liquid effluent pathways and the cooling tower blowdown have flow measuring devices.

460.18 (SRP 11.5 & II.F.1, NUREG-0737) With regard to TMI Action Plan II.F.1, Attachments 1 and 2, of NUREG-0737, November 1980, "Clarification of TMI Action Plan Requirements," you have not provided adequate information.

For Attachment 1 of II.F.1, "Additional Accident Monitoring Instrumentation - Noble Gas Effluent Monitor," provide information on:

a. items covered in Clarification No. 4 (page II.F.1-3, NUREG-0737); and
b. sampling design criteria, power supply and method of calibration (see Table II.F.1-1 of NUREG-0737).

For Attachment 2, "Additional Accident Monitoring Instrumentation - Sampling and Analysis of Plant Effluents":

a. include the north and south plant vent stacks (your current writeup on page 1.10-60 of the HCGS FSAR considers only the FRVS); and
b. provide information on 1) design basis shielding envelope, 2) efficiencies for capture of radioiodine and particulate by the sampling media, 3) nature of the samplers, i.e., whether they are silver zeolite cartridges etc., and 4) corrections that you will make for non-isokinetic sampling conditions (for all these items, see Table II.F.1-2 of NUREG-0737).

Note that your information on the TMI Action Plan II.F.1 of NUREG-0737 should also include a commitment to perform the required human-factor analysis stated on page II.F.1-1 of NUREG-0737.

460.19 (SRP 15.7.3) HCGS FSAR Subsections 2.4.13.3 and 2.4.13.4 do not adequately address the off-site radiological consequences resulting from postulated radioactive releases due to liquid tank failures. Therefore, provide an analysis for the off-site radiological consequences due to postulated liquid tank failures, using the assumptions given in SRP 15.7.3, Rev. 2, July 1981, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures."