JUL 24 1984

Docket No. 50-354

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

FACILITY: Hope Creek Generating Station

SUBJECT: SUMMARY OF CHEMICAL ENGINEERING BRANCH DRAFT SER OPEN ITEM MEETING

On July 13, 1984, a meeting was held in the Bethesda, Maryland offices of the NRC to discuss open items identified in the Draft SER by the Chemical Engineering Branch (Chemical Technology Section). A list of attendees is included as Enclosure 1.

The items discussed, and their status are indicated in Enclosure 2. PSE&G was requested to provide revised formal responses to the enclosed items by July 31, 1984.

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Licensing Branch No. 2
Division of Licensing

Enclosures: As stated

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

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Enclosure 1

Chemical Engineering Branch Meeting

Name	Affiliation
Dave Wagner	NRC
Stan Kirslis	NRC
Frank Witt	NRC
William Gailey	PSE&G
Bruce Preston	PSE&G
Jim Ashley	PSE&G
J.C. Bradford	Bechtel
Ronald Drewnowski	PSE&G
D.B. James	GE
Bernard Turvolin	NRC

ENCLOSURE 2

DSER Section 9.1.2

Additionally, the information provided through Amendment 3 was not sufficient for the staff to complete the evaluation of the compatibility and chemical stability of materials wetted by spent fuel pool water. To complete the review, the following information is requested:

(1) Identify and list all materials in the spent fuel storage pool including the neutron poison material, rack leveling feet, and rack frame.

(2) Provide test or operating data showing that the neutron poison material will not degrade during the lifetime of the spent fuel storage pool.

(3) Provide a description of any materials monitoring program for the pool. In particular, provide information on the frequency of inspection and type of samples used in the monitoring program.

(4) Provide details of the spent fuel racks to show that no buildup of gases will occur in the cavities containing the poison materials.

Pending receipt and review of this information, this is an open item.

STATUS

The attached response was originally provided at the Auxiliary Systems Branch meeting of May 30 and 31, 1984. At the Chemical Engineering Branch meeting, the NRC reviewer noted that the following information should be provided:

1. a copy of the "boral" report or a reference to the report
2. identification of plants that use boral clad with stainless steel
3. drawings showing detail of the upper edges of the racks (to assure venting of gasses)
4. assurance that scale is removed from adjusting screw materials

DSER Open Item No. 140 (DSER Section 9.1.2)

SPENT FUEL STORAGE

Since the applicant's application for an operating license was docketed in 1983, which is after the November 17, 1977 date specified in the SRP, the applicant must provide the results of an analysis which shows that a failure of the liner plate as a result of an SSE will not cause any of the following: (1) significant releases of radioactivity due to mechanical damage to the fuel; (2) significant loss-of-water from the pool which could uncover the fuel and lead to release of radioactivity due to heat up; (3) loss of the ability to cool the fuel due to flow blockage caused by a portion of one or more complete section of the liner plate falling on the top of the fuel racks; (4) damage to safety-related equipment as a result of the pool leakage; and (5) uncontrolled release of significant quantities on radioactive fluids to the environs; in accordance to the Standard Review Plan. These buildings are also designed against flooding and tornado missiles (refer to Section 3.4.1 and 3.5.2 of this SER). We cannot conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Position C.3, 1.29, "Seismic Design Classification," Positions C.1 and C.2, have been met.

The applicant has not provided the design details of the spent fuel storage racks, the results of an analysis of impacts onto the racks, the bundle to bundle spacing, the design maximum enrichment (weight percent of U235), a description of calculational methods used for criticality analysis (along with the results), a tabulation of the nominal value of K_{eff} of the racks along with the various uncertainties and biases considered in the analysis, and a tabulation of the reactivity effect of each of the abnormal accident situations considered for our review. Since credit is taken for gadolinia in the fuel, the applicant must provide a commitment that every fuel bundle will have a specified minimum amount of gadolinia distributed over a specified number of specific fuel pins, for the entire length of the fuel. As an alternative, the applicant can provide the results of the criticality analysis without taking credit for the gadolinia.

Thus, we cannot conclude that the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13, Positions C.1 and C.4, concerning fuel storage facility design are satisfied.

DSER Open Item No. 140 (Cont'd)

We cannot conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 61, and 62 as they relate to protection of the spent fuel against natural phenomena, radiation protection, and prevention of criticality and the guidelines of Regulatory Guides 1.13, Positions C.1, C.3, and C.4 and 1.29, Positions C.1 and C.2, relating to the facility's design basis and seismic classification. The spent fuel storage facility does not meet the acceptance criteria of SRP Section 9.1.2. We will report resolution of this item in a supplement to this SER.

Additionally, the information provided through Amendment 3 was not sufficient for the staff to complete the evaluation of the compatibility and chemical stability of materials wetted by spent fuel pool water. To complete the review, the following information is requested:

(1) Identify and list all materials in the spent fuel storage pool including the neutron poison material, rack leveling feet, and rack frame.

(2) Provide test or operating data showing that the neutron poison material will not degrade during the lifetime of the spent fuel storage pool.

(3) Provide a description of any materials monitoring program for the pool. In particular, provide information on the frequency of inspection and type of samples used in the monitoring program.

(4) Provide details of the spent fuel racks to show that no buildup of gases will occur in the cavities containing the poison materials.

RESPONSE

The spent fuel pool liner plate was not designed to seismic Category I requirements because SRP 9.1.2, Revision 2 (March 1979), which first invoked the seismic Category I requirement, was not issued until after the design and procurement of the liner plate was complete and fabrication had begun (November 1978). However, the liner plate was designed to act as a form for the concrete in the spent fuel pool walls. To perform this function a system of channels, wide flanges and angle stiffeners was welded to the back surfaces of the liner and connected to the outside formwork with form ties. Thus, during the concrete pouring operation the welds between the stiffeners and the liner were subject to the lateral pressure effects of the wet concrete. This may be considered a 'test' load in that after the concrete sets, the anchoring capability

RESPONSE (Cont'd)

of the stiffener system in holding the liner plate against seismic loads is at least equal to the form pressure load. The estimated test during construction (approximately 300 lb/ft^2) was lower than the design value of 690 lb/ft^2. This construction load induced a correspondingly lower stress in the stiffener-to-liner welds.

An analysis, performed to evaluate the effect of SSE loads on the liner, shows that the resultant stresses would be insignificant (approximately 1% of the stresses due to concrete placement) when added to the residual concrete load.

Thus, the design of the liner plate satisfies General Design Criteria 2, 61, and 62, Regulatory Guide 1.29, Positions C.1 and C.2, and Regulatory Guide 1.13, Positions C.1 and C.4. Refer to Section 9.1.2.5 for additional justification of the non-seismic Category I liner design. For additional information on the design and analysis of the liner plate, refer to Appendix 3F.

For a discussion of the liner leakage collection system, which permits expedient liner leak detection and measurement, and prevents uncontrolled loss of contaminated pool water, refer to Section 9.1.2.2.2.1.

The spent fuel storage facility design meets the intent of Regulatory Guide 1.13 Position C.3, as described in Section 9.1.4.6 and 9.1.5.6.

The spent fuel storage rack design details will be provided in the response to Questions 220.15, 281.2, 281.13, 410.38, 410.39 and 410.42. The requested information will be available by July, 1984, and will be provided by August, 1984.

The materials used in the spent fuel storage racks were included in the response to Question 281.13 (Amendment 5).

Similar rack designs, with vented Boral poison in stainless steel racks, have been licensed and have proven successful. The Yankee Rowe racks have been in use since 1964 without a Boral poison failure. NRC sponsored tests at Brookhaven National Lab support the use of Boral poison material. Brooks and Perkins Product Performance Report 624 provides additional justification and is available for NRC review.

In order to continually assure the adequacy of the poison material, test coupons are provided for a Boral surveillance program. Forty-five coupons are installed in high radiation areas of the spent fuel pool. However, because stainless steel spent fuel racks with Boral poison material are already in use in other BWR fuel pools, a Boral surveillance program is not planned at HCGS.

RESPONSE (Cont'd)

If information from these lead plants indicates any problem with the Boral, a surveillance program can then be initiated.

The spent fuel rack poison cavities are vented to prevent any buildup of gases. Response to Question 281.13 for provides further information on venting.

QUESTION 281.13 (SECTION 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.

RESPONSE

All parts of the spent fuel racks, except the adjusting screws in the feet of each module and the poison material, are made from ASTM A240, Type 304L, stainless steel. The adjusting screws are made from ASTM A564, Type 630 stainless steel. Boral is the poison material.

Thin (0.024 inch thick) outer canister sheets hold the Boral tightly against the 0.090 inch thick inner canister walls. The outer canisters are spot welded to the inner canisters along the bottom and both vertical sides of the outer canister. The top edge of each outer canister is seam welded to the inner canister. The gaps between the spot welds provide the poison venting.

DSER Section 9.1.3

Additionally the information provided through Amendment 3 was not sufficient for the staff to complete its evaluation of the spent fuel pool sampling and monitoring. To complete the review, the following information is needed:

(1) Describe the sampling procedure, analytical instrumentation, and sampling frequency for monitoring spent fuel pool purity.

(2) State the radiochemical limits for initiating corrective action.

The applicant's response should consider permissible gross gamma and iodine activities and the demineralizer decontamination factor.

Pending receipt and review of this information, this is an open item.

STATUS

The attached response was provided by letter dated June 15, 1984 (R. L. Mittl to Schwencer). Based on the review of the response, the applicant needs to provide detail concerning the radiochemical limits for initiating corrective actions.

DSER Open Item No. 141g (Section 9.1.3)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Additionally, the information provided through Amendment 3 was not sufficient for the staff to complete its evaluation of the spent fuel pool sampling and monitoring. To complete the review, the following information is needed:

(1) Describe the sampling procedure, analytical instrumentation, and sampling frequency for monitoring spent fuel pool purity.

(2) State the radiochemical limits for initiating corrective action.

The applicant's response should consider permissible gross gamma and iodine activities and the demineralizer decontamination factor.

RESPONSE

FSAR Section 9.1.3.2.2.4 has been revised to provide the requested information.

The stainless steel filter-demineralizer vessels are of the pressure precoat type. A tube nest assembly consisting of the tube sheet, clamping plate, filter elements, and support grid is inserted as a unit between the flanges of the vessel. The filter elements are stainless steel and are mounted vertically in the vessel. Air scour connections are provided below the tube sheet, and vents are provided in the upper head of each vessel. The filter elements are installed and removed through the top of each vessel. The holding elements are designed to be coated with powdered ion exchange resin as the filtering medium.

The fuel pool filter-demineralizers maintain the following effluent water quality specifications:

Specific conductivity at 25°C, micromho/cm	≤0.1
pH at 25°C	6.0 to 7.5
Heavy elements (Fe, Hg, Cu, Ni), ppm	0.05
Silica (as SiO_2), ppm	<0.05
Chloride (as Cl-), ppm	<0.02
Total insolubles, ppm	90% removal to a minimum of 0.01 ppm

Insert A →

The filter-demineralizers are designed to be backwashed periodically with water to remove resin and accumulated sludge from the holding elements. Service air pressure loosens the material from the holding elements and the backwash slurry drains through the gravity drainline to the waste sludge phase separator in the solid waste management system.

The resin tank provides adequate volume for one precoating of one filter demineralizer vessel.

The resin eductor transfers the precoat mixture of resin to the holding pump suction line at a flow rate of 4 gpm.

The holding pumps are designed to recirculate a uniform mixture of resin through the filter-demineralizer vessel being precoated at a flow rate of 1.5 gpm/ft² of filter element surface area, and to automatically start and maintain the precoat material on the filter elements when the system flow rate falls below the value necessary to keep the precoat on the elements.

INSERT A

The influent and effluent water of the FPCC is continuously monitored by on line PH and conductivity instrumentation. In addition, grab samples of the influent H_2O will be analyzed once per week for chloride and for gamma isotopic and once per month for heavy metals (Fe, Cu, Hg, Ni). Grab samples of effluent water will be analyzed weekly for chloride, silica, suspended solids, tritium, and for gamma isotopes.

Decontamination factors (df) of greater than 10 are expected for any chloride present and greater than 5 for isotopes of Iodine and Cobalt. Resin bed(s) will be regenerated and/or replaced when these dfs are not achieved.

DSER OPEN ITEM 141g

M P84 95/05 3-dh

DSER Section 9.3.2

Postaccident Sampling System, TMI - 2 Action Plan Item II.B.3

The information provided through Amendment 3 was not sufficient for the staff to complete its evaluation. This is an open item.

To meet the criteria of NUREG-0737, Item II.B.3, the guidelines of Appendix B to this SER should be implemented.

STATUS

The attached proposed response was provided at the meeting. Based on staff review of the resonse and discussions at the meeting, the following revisions to the response will be made:

(a.4)	1.	PSE&G should state intentions of using a licensed shipping cask
(c)	2.	response will state that valves not accessible after an accident can perform their function in an accident environment
(f)	3.	PSE&G needs to supply a response on GDC-19 with regard to the PASS
(i)	4.	PSE&G will clarify the response
(j)	5.	sensitivities of onsite sampling instruments will be provided in tabular form.
(k)	6.	PSE&G will insert in response a statement saying the containment atmosphere sample line is heat traced

DSER Open Items No. 148 (DSER Section 9.3.2)

Postaccident Sampling System, TMI-2 Action Plan Item II.B.3

The information provided through Amendment 3 was not sufficient for the staff to complete its evaluation. This is an open item.

To meet the criteria of NUREG-0737, Item II.B.3, the guidelines of Appendix C to this SER should be implemented.

RESPONSE

See attached marked-up response to Question 281.15.

QUESTION 281.15 (SECTION 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.

RESPONSE

Section 9.3.2 has been revised to provide the information responding to the attachment transmitted with this question.

Additional information on the following will be provided in June 1984.

- Equipment used to ship samples for offsite analyses
- Time to analyze samples
- Quantification methods
- Chloride analysis
- Compliance with GDC19 for PASS sample analysis

The attached FSAR pages respond to the above items, except compliance to GDC19 which will be provided by September 1984.

However, NUREG 0737 Section II.B.3.1 requires that the PASS meet the following:

a. 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.

The following is a conservative time sequence for sampling, transport, and analysis to demonstrate that samples can be obtained and analyzed within the specified 3-hour period:

1. Recirculate sample, install sample vial/or cartridge -- 15 min.

2. Operate sample station -- 15 min.

3. Transport sample to lab -- 20 min.

ADD

4. Analyze sample - 30 min.

Sample points and sample gathering methods are discussed in Section 9.3.2.2.2.

b. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

1. 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 nonvolatile isotopes);

2. Hydrogen levels in the containment atmosphere;

A chloride analysis will need to be performed within 4 days of the sample being taken because 1) the plant has brackish coolant water and 2) two barriers are provided between primary containment systems and the cooling water (see Figure 9.2-3).

f. 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).)

The PASS radiation shielding design will be in accordance with Section 12.3.2.2.6 to keep personnel exposures as low as practicable and within the limits established by GDC 19.

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

ADD INSERT "A" (2A)

h. If inline monitoring is 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 samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per week until the accident condition no longer exists.

The Hope Creek Generating Station PASS is a grab sample system and is described in Section 9.3.2.2.2.

INSERT "A"

(procedure, CH-No. CA.22-25)

HCGS will develop a procedure for Boron analysis prior to core load.

A chloride analysis will need to be performed within 4 days of the sample being taken because 1) the plant has brackish coolant water and 2) two barriers are provided between primary containment systems and the cooling water (see Figure 9.2-3).

f. 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).)

The PASS radiation shielding design will be in accordance with Section 12.3.2.2.6 to keep personnel exposures as low as practicable and within the limits established by GDC 19.

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

h. If inline monitoring is 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 samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per week until the accident condition no longer exists.

ADD INSERT 'B'

~~The Hope Creek Generating Station PASS is a grab sample system and is described in Section 9.3.2.2.2.~~

Delete

INSERT 'B'

HCGS PASS has the capability to obtain grab samples as described in Section 9.3.2.2.2.

i. The licensee's radiological and chemical sample analysis capability shall include provisions to:

1. 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 1 μCi/g to 10 Ci/g.

2. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptable 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 ventilation system design which will control the presence of airborne radioactivity.

A diluted liquid sample can be obtained as described in Section 9.3.2.2.2.6.

ADD INSERT C. (2)

j. 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.

k. In the design of the postaccident and analysis capability, consideration should be given to the following items:

1. 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 postaccident reactor coolant and

INSERT C

All sample bottles, iodine cartridges, etc., will be identified prior to sampling to eliminate unnecessary exposure resulting from handling high level samples. A centralized logging system will be developed to track sample aliquot identification, dilution factors, sample disposition, etc.

Liquid samples will be taken at the sample station in septum-type bottles and trahsported to the analysis facility in lead containers.

Sample aliquots are taken from the septum bottles for analysis or further dilution. Aliquoting and transfer will be performed using shielded containers, or behind a lead brick pile. Calibrated hypodermic syringes will be used for aliquoting the higher activity samples. Tongs or other holding/clasping devices will be available for holding the sample bottle during the transfer and dilutions to reduce hand and body exposure. Unless prohibited by the intended analysis, dilutions will be done using very dilute (about 0.01N) nitric acid as the diluent to minimize sample plateout problems.

Primary coolant samples obtained from the sampling station are diluted by a factor of 100 (0.1 ml coolant diluted to 10 ml). Under severe accident conditions, a calibrated syringe would be used to obtain an aliquot for this sample for further dilutions. At the maximum expected primary coolant activity level (3 Ci/cc), a dilution factor of 1×10^5 would be required for gamma spectroscopy.

Direct counting of the initial 100:1 dilution sample would allow analysis at coolant activity levels down to 1 Ci/cc. In addition, the degassed, undiluted 10 ml sample available from the sample station could be used for analysis of samples in the 10^{-2} to 10^{-3} Ci/cc range. Thus, useful samples may be obtained from the post-accident sampling station for coolant activity levels ranging from design basis accident source terms to well below the maximum level that can be tolerated at the normal reactor sample station.

However, NUREG 0737 Section II.B.3.1 requires that the PASS meet the following:

a. 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.

The following is a conservative time sequence for sampling, transport, and analysis to demonstrate that samples can be obtained and analyzed within the specified 3-hour period:

1. Recirculate sample, install sample vial/or cartridge -- 15 min.

2. Operate sample station -- 15 min.

3. Transport sample to lab -- 20 min.

4. Analyze sample -

Sample points and sample gathering methods are discussed in Section 9.3.2.2.2.

b. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

1. 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 nonvolatile isotopes);

ADD INSERT 'A' (2X)

2. Hydrogen levels in the containment atmosphere;

INSERT "A"

A generic procedure to assess the extent of core damage based on radionuclide concentrations and other parameters has been prepared by the BWR Owners Group (see Chapter 1.8.1.9.7). A HCGS plant specific procedure based on this methodology will be prepared by 1/85.

However, NUREG 0737 Section II.B.3.1 requires that the PASS meet the following:

a. 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.

The following is a conservative time sequence for sampling, transport, and analysis to demonstrate that samples can be obtained and analyzed within the specified 3-hour period:

1. Recirculate sample, install sample vial/or cartridge -- 15 min.

2. Operate sample station -- 15 min.

3. Transport sample to lab -- 20 min.

4. Analyze sample -

Sample points and sample gathering methods are discussed in Section 9.3.2.2.2.

b. The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

1. 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 nonvolatile isotopes);

2. Hydrogen levels in the containment atmosphere;

ADD INSERT "B" (23)

INSERT 'B'

At greater than 15% power, the primary containment atomsphere is maintained under a nitrogen blacket. Hydrogen and oxygen concentrations are monitored by chemical analysis of gas samples drawn from various points in the drywell and torus. During post accident conditions, hydrogen and oxygen concentrations are monitored by one of two Hydrogen/Oxygen analyzers. ~~Concentrations are maintained at less than 6% hydrogen and 4% oxygen (by volume) by venting through the Reactor Building Ventilation System and/or providing nitrogen makeup as required~~.

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

ADD* INSERT 'C' (24)

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

Inline monitoring capabilities (radiation monitors and conductivity cell) are discussed in Section 9.3.2.5.2.

c. Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

Isolated auxiliary systems are not required for PASS operation. The PASS is described in Section 9.3.2.2.2.

d. 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.

The method of gathering pressurized and non-pressurized reqactor coolant samples is discussed in Section 9.3.2.2.2.

e. The time for a chloride analysis to be performed is dependent 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.

Insert "C"

Total Dissolved Gas analysis will be perfromed by the method recommended by the BWR Owners Group and GE (as discussed in Section 1.8.1.97).

Chloride analysis will be performed by Ion Chromatography; Boron by Specific Ion Electrode.

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

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

ADD INLINE D.

Inline monitoring capabilities (radiation monitors and conductivity cell) are discussed in Section 9.3.2.5.2.

c. Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

Isolated auxiliary systems are not required for PASS operation. The PASS is described in Section 9.3.2.2.2.

d. 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.

The method of gathering pressurized and non-pressurized reqactor coolant samples is discussed in Section 9.3.2.2.2.

e. The time for a chloride analysis to be performed is dependent 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.

Insert "D"

HCGS will have the capability of sending samples offsite. Arrangements will be made with offsite facilities to perfrom analyses and an appropriate shipping cask will be obtained prior to core load.

The small volume (diluted) liquid sample cask is a cylinder with a lead wall thickness of about 2 inches. The cask weighs approximately 50 pounds and has a handle which allows it to be carried by one person.

The 10 milliliter undiluted sample is taken in a 700 pound lead shielded cask which is transported and positioned by a four-wheel dolly. The sample is shielded by about 5-1/2 inches of lead.

9.3.2.2.2.10 PASS Power Supply

The PASS isolation and control valves, sample station control panels, isolation valve control panels, and auxiliary equipment are connected to a non-1E battery backed power source. The safety auxiliaries cooling system, which is needed for the sample coolers, is powered from the emergency diesel generators following a loss of offsite power. Power for the gas sample line heat tracing is supplied from a diesel backed source.

9.3.2.3 Safety Evaluations

9.3.2.3.1 Process Sampling System Safety Evaluation

The process sampling system has no safety-related function. Failure of the system will not compromise any safety-related system or component, or prevent a safe shutdown of the plant.

The process sampling lines, connected to the reactor coolant pressure boundary (RCPB) through the first isolation valve outside containment, are designed to seismic category I requirements, as defined in Section 3.7. Sample lines that penetrate the containment are provided with isolation valves in accordance with 10 CFR 50, Appendix A, GDC 55, as described in Section 6.2.4.

9.3.2.3.2 Post-Accident Sampling System Safety Evaluation

The PASS has no safety-related function. Failure of the system will not compromise any safety-related system or component, or prevent a safe shutdown of the plant.

INSERT 'A'

9.3.2.2.2.11 Storage and Disposal of Sample

Short-term sample storage areas will be provided in the chemistry laboratory and counting room facilities. An area for long-term storage of the samples will be designated prior to core load. Low level wastes generated by routine chemistry evolutions will be flushed to radwaste. Procedures addressing the ultimate disposal of the samples will be provided by 1/85.

1.8.1.97 Conformance to Regulatory Guide 1.97, Revision 2, December 1980: Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

HCGS complies with the BWR Owner's Group position (Reference 1.8-4) on Regulatory Guide 1.97 with the following clarifications and exceptions:

a. Suppression chamber spray flow (Type D variable) - The BWR Owner's Group has recommended not implementing this variable. HCGS has implemented this variable as Category 2.

b. Drywell spray flow (Type D variable) - The BWR Owner's Group has recommended not implementing this variable. HCGS has implemented this variable as Category 2.

c. Condenser cooling water flow (BWR Owner's Group recommended Type D variable) - HCGS deviates from the BWR Owner's Group position on this variable by using the cooling water temperature rise (delta T) across the condenser to provide this information.

VSECT A

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

1.8.1.98 Conformance to Regulatory Guide 1.98, Revision 0, March 1976: Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

HCGS complies with Branch Technical Position ETSB 11-5, Revision 0, July 1981, in lieu of Regulatory Guide 1.18.

For further discussion, see Section 15.7.1.

d. Total Dissolved Gas Analysis - The guidelines recommended by the BWR Owners Group and GE shall be followed. This was agreed to in a meeting between NRC Management (R. Vollmer, et. al) and GE (F. Quirk, et. al), dated December 12, 1983.

i. The licensee's radiological and chemical sample analysis capability shall include provisions to:

1. 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 1 μCi/g to 10 Ci/g.

2. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptable 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 ventilation system design which will control the presence of airborne radioactivity.

A diluted liquid sample can be obtained as described in Section 9.3.2.2.2.6.

j. 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.

Add Item A. (24)

k. In the design of the postaccident and analysis capability, consideration should be given to the following items:

1. 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 postaccident reactor coolant and

HCGS

Insert A

1. On-site chloride will be determined by Ion Chromatography.

2. A combination electrode will be used to measure the pH of coolant samples. Testing performed by GE has verified that expected levels of irradiation result in a shift of less than 0.3 pH units.

3. The boron determination is made on a 1:100 dilution of reactor water.

4. The post-accident sample station is equipped with a 0.1 uS conductivity cell. The conductivity meter has a linear scale with a six-position range of 0-3, 0-10, 0-30, 0-100, 0-300 and 0-1000 uS when using the 0.1 uS cell.

   Conductivity measurements are, of course, non-specific, but they serve the important function of indicating changes in chemical concentrations and conditions. Perhaps even more important, in the case of the BWR primary coolant, the conductivity measurements can establish upper limits of possible chemical concentrations and can eliminate the need for additional analyses.

   The conductivity measurement can also be used to bound the possible range of pH values.

5. Equipment used for post-accident sampling and analysis will be calibrated or tested approximately every six months. Personnel classroom training in the collection and analysis of samples will be performed every six months.