ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report Nos. 50-317/85-16
50-318/85-14

Docket Nos. 50-317
50-318

License Nos. DPR-53 Category C
DPR-69

Licensee: Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Facility Name: Calvert Cliff Nuclear Power Plant, Units 1 and 2

Inspection At: Lusby, Maryland

Inspection Conducted: June 24 - 28, 1985

Inspectors: J. R. White 8/5/85
J. R. White, Senior Radiation Specialist date

for A. A. Weadock, Radiation Specialist 8/5/85 date

for L. S. Cheung, Reactor Engineer 8/5/85 date

for P. E. Clemons, Radiation Specialist 8/5/85 date

for W. Knox, Contractor, Brookhaven National Laboratory 8/5/85 date

for E. Lessard, Contractor, Brookhaven National Laboratory 8/5/85 date

Approved by: M. Shanbaky 8/6/1985
M. M. Shanbaky, Chief, PWR Radiation Safety Section date

Inspection Summary:
Inspection on June 24 - 28, 1985 (Report Nos. 50-317/85-16; 50-318/85-14)

Areas Inspected: Special, announced safety inspection of the licensee's implementation and status of the following task actions identified in NUREG-0737: II.B.3, Post-accident sampling of reactor coolant and containment atmosphere; II.F.1-1, Noble gas effluent monitors; II.F.1-2, Post-accident effluent monitoring; II.F.1-3, Containment radiation monitoring; and, III.D.3.3, In-plant radioiodine measurements. The inspection involved 190 hours by four region-based inspectors and two contractors from Brookhaven National Laboratory.

Results: Several deficiencies were identified. The following deficiencies appear to represent violations of NRC requirements: Failure to comply with the Limiting Condition of Operation specified by Technical Specification 3.7.13, "Post Accident Sampling"; Failure to assure environmental qualification of the Containment High Radiation Monitors pursuant to NUREG-0737 as confirmed by the NRC Confirmatory Order dated March 14, 1983; Failure to perform surveillances of gasecus effluent monitoring instrumentation used for post-accident monitoring pursuant to Technical Specification 4.3.3.8, "Radiation Gaseous Effluent Monitoring Instrumentation"; and, failure to implement a personnel training program pursuant to the requirements of Technical Specification 6.15, "Iodine Monitoring." Additionally, licensee had not implemented and maintained the post-accident sampling system with respect to submittals that were confirmed by an NRC Confirmatory Order, dated March 14, 1983.

DETAILS

1.0 Persons Contacted

*J. A. Tiernan	Manager, Nuclear Power, Baltimore Gas & Electric (BG&E)
*P. G. Rizzo	Supervisor, Technical Training, Calvert Cliffs Nuclear Power Plant (CCNPP)
*R. E. Denton	General Supervisor, Training, CCNPP/BG&E
*R. L. Wenderlich	Supervisor, Operations Quality Assurance Auditing
*L. E. Salyards	Senior Engineer - Licensing, BG&E
*M. J. Miernicki	Principal Engineer - Licensing, BG&E
*G. F. Wall	Engineering Analyst. CCNPP/BG&E
*C. L. Rayburn	Emergency Planning Analyst, CCNPP/BG&E
*R. B. Sydnon	Supervisor - Electrical and Control, CCNPP/BG&E
*N. L. Millis	General Supervisor - Radiation Safety, CCNPP/BG&E
*P. T. Crinigan	General Supervisor - Chemistry, CCNPP/BG&E
*B. N. Proctor	Technical Support Engineer, CCNPP/BG&E
*G. C. Wolf	Technical Support Engineer, CCNPP/BG&E
A. Marion	Senior Engineer - Electrical Engineering, BG&E

*Denotes attendance at the exit interview conducted on June 28, 1985.

Other members of the licensee's staff were also contacted and/or participated in exercises of the post-accident sampling and the effluent monitoring systems during the inspection.

2.0 Purpose

The purpose of this inspection was to verify and validate the adequacy of the licensee's implementation of the following task actions identified in NUREG-0737, Clarification of TMI Action Plan Requirements:

Task No.	Title
II.B.3.	Post Accident Sampling Capability
II.F.1-1	Noble Gas Effluent Monitors
II.F.1-2	Sampling and Analysis of Plant Effluents
II.F.1-3	Containment High-Range Radiation Monitor
III.D.3.3	Improved Inplant Iodine Instrumentation under Accident Conditions

3.0 Executive Summary

The following summary is a overview of the most significant findings of this inspection.

3.1 Post Accident Sampling Capability, Item II.B.3

This review indicated that while the licensee considered the system to be installed June 1, 1983, pursuant to commitments contained in

the NRC "Order Confirming Licensee Commitments on Post-TMI Related Issues," dated March 14, 1983, the item was not implemented and maintained in accordance with those commitments, in that the system was never demonstrated nor could it function as described in submittals to the NRC.

At the time of this inspection, in-line analytic components (i.e, Boron Analyzer and pH Analyzer) were inoperable; and other equipment (i.e., Radioisotopic Analyzer and Hydrogen/Oxygen Analyzer) still remained to be demonstrated as able to function as specified in submittals to the NRC. Additionally, it was determined that certain valves necessary to establish sample flow through the system would not operate during a system demonstration; and the system's ability to provide diluted grab samples for backup analysis was not reliable since a known dilution factor could not be verified.

Upon issuance of a specific Technical Specification referencing post-accident sampling on February 22, 1985 (i.e., Section 3/4.7.13) the licensee did document the systems continued inoperability, indicating that the preplanned alternate method of processing samples was in effect.

This inspection determined that the licensee was utilizing a sampling technique involving the station's routine sample sink and post-accident sampling apparatus originally developed to meet interim requirements of NUREG-0578 to meet the backup sampling capability requirements of NUREG-0737. This sampling technique had not been submitted to or evaluated by the Commission as to its adequacy in fulfilling the requirements of NUREG-0737, Item II.B.3. Review of this sampling method revealed that it was not preplanned, in that procedures were not commensurate with actual system configuration, nor were personnel trained in the method; and it was not a practicable sampling scheme for post-accident conditions in that it likely involved incurring personnel exposure in excess of GDC-19 criteria and could not be demonstrated as a workable solution to post-accident sampling.

3.2 Noble Gas Effluent Monitors, Item II.F.1-1

The licensee is using two systems, the Wide Range Gas Monitoring (WRGM) system and the Main Steam Effluent Radiation Monitor (MSERM) system to meet the noble gas effluent pathway monitoring requirements of NUREG-0737 for noble gas monitoring; however, the following concerns were identified during this inspection:

The following concerns were identified relative to the WRGM system:

- A spare parts inventory for timely system repair is not maintained by the licensee.
- Problems with vent stack flow instrumentation require the use of a default flow rate value.

- The majority of required surveillance and maintenance procedures for this system have not been developed. Consequently, formalized training in these procedures has not been given.

The MSERM system will be used to monitor noble gas releases to the atmosphere through the main steam line pathway. This system has not been declared operational by the licensee. Equipment has been installed but final calibrations have not been completed. Licensee commitments require operability for the Unit 1 system by the end of the current outage and by December 31, 1985, for the Unit 2 system.

The following concerns were identified relative to the MSERM system:

- Formal procedures and training controlling the operation and upkeep of this system have not been developed.

- Calibration data showing monitor response to noble gas activity (in μCi/cc) was not available.

- Information was not available demonstrating that the attenuation of low range gammas by main steam piping had been considered in determining detector response.

3.3 Sampling and Analysis of Plant Effluents, Item II.F.1-2

The licensee is utilizing the grab sampling capability of the Wide Range Gas Monitor (WRGM) system to meet the radioiodine and particulate effluent sampling requirements of NUREG-0737. The WRGM system allows diversion of the main vent stack sample stream through shielded, quick disconnect particulate and iodine filters. These filters can then be manually transported to the filter analysis point.

Based on a review of system capabilities the WRGM system was determined to be unacceptable in meeting the sampling requirements of NUREG-0737, Item II.F.1-2. Of primary concern is the failure of the licensee to demonstrate the system is providing representative iodine and particulate sampling; and failure to perform operability surveillance of the system in accordance with technical specifications. Other concerns identified during this inspection include:

- Failure to address adequacy of the installed sample line heat tracing to provide adequate heating under all ambient temperature conditions.

- Failure to provide adequate procedures and personnel training for filter removal, handling, and subsequent analysis.

- Failure to perform an adequate time and motion study for filter retrieval and analysis which takes into account all radiation sources.

3.4 Containment High Range Radiation Monitor, Item II.F.1-3

The licensee's implementation of this NUREG-0737 requirement generally appeared to be in accord with the specifications. Since this system is monitoring inside of containment and is expected to function in accident conditions, such as LOCA, the installation was specified to be environmentally qualified by NUREG-0737.

The licensee indicated in submittals to the NRC that the system was installed pursuant to NUREG-0737 requirements. An NRC Confirmatory Order documented this commitment. However, direct observation of the installation in Unit 1 revealed that certain protective sleeving necessary to assure environmental qualification of the monitors' electrical connectors at the internal containment penetrations were not installed due to a maintenance oversight. Such lack of protective sleeving would compromise the system's operation in accident environments.

3.5 Inplant Radioiodine Monitoring, Item III.D.3.3

The licensee's implementation of this NUREG-0737 requirement generally appeared to be in accord with the specifications. The licensee's Technical Specification 6.15 requires the implementation of a personnel training program for monitoring radioiodine. Such a program is defined in the licensee's procedures which specifies a yearly requirement for training.

However, at the time of this inspection it was found that no personnel have been trained in this area since February 1984, indicating that the licensee has failed to implement the program as specified by procedures.

4.0 TMI Action Plan Generic Criteria and Commitments

The licensee's implementation of the task actions specified in Section 2.0 were reviewed against criteria and commitments contained in the following documents:

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

- Letter from D. G. Eisenhut, Acting Director, Division of Operating Reactors, to all Operating Power Plants, dated October 30, 1979.

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

- Generic Letter 82-05, letter from D. G. Eisenhut, Director, Division of Licensing, to All Licensees of Operating Power Reactors, dated March 14, 1982.
- NRC "Order Confirming Licensee Commitments on Post-TMI Related Issues", dated March 14, 1983.
- Regulatory Guide 1.4, "Assumptions Used for Evaluating Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors".
- Regulatory Guide 1.97, Rev. 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident".
- Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Station will be As Low As Reasonably Achievable".

5.0 Post Accident Sampling System, Item II.B.3.

5.1 Position

NUREG-0737, Item II.B.3, specifies that licensees shall have the capability to promptly collect, handle and analyze post-accident samples which are representative of conditions existing in the reactor coolant and containment atmosphere. Specific criteria are denoted in commitments to the NRC relative to the specifications contained in NUREG-0737.

Documents Reviewed

The implementation, adequacy and status of the licensee's post-accident sampling and monitoring systems were reviewed against the criteria identified in Section 4.0 of this report and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment I.A.

5.2 System Description

The Calvert Cliffs Post Accident Sampling System (CE-PASS) was designed and built by Combustion Engineering. The system is designed to permit in-line analysis of the chemical and isotopic content of reactor coolant. The system also has provisions for the collection of diluted reactor coolant samples for laboratory analysis.

The CE-PASS is common to both units. It consists of a control panel, a sampling station, a shielded germanium radiation detector for isotopic analysis, signal processing and power supply panels, in-line analyzers for boron, pH, and dissolved gas (hydrogen and oxygen) analyses, and valve operators. These components are

all physically located in the Solid Waste Handling Room on the 45-foot level of the Auxiliary Building.

From the control panel a chemistry technician can remotely control and monitor in-line analysis of post-accident reactor coolant, and quantify the sample with respect to hydrogen, oxygen, boron and pH. Additionally, in-line radioisotopic analysis is provided to qualify and quantify various radioisotopes in the reactor coolant. Chloride determinations are made from a diluted grab sample provided by the CE-PASS.

The signal processing panel contains the electronic circuitry needed to process the germanium detector signals for subsequent interpretation at the chemistry laboratory's multi-channel analyzer. The power supply panels contain devices used to provide regulated power to the detector and signal processing circuits.

Figure 1, "Simplified Drawing-CE-PASS." depicts the general arrangement of the system. It is designed to collect and to analyze reactor coolant samples from:

a. the hot leg directly, at operating pressure;

b. the hot leg via Low Pressure Safety Injection (LPSI), at low pressure; and

c. the containment sump, via the LPSI.

At the time of this inspection, the licensee considered backup sampling and analysis capability required by NUREG-0737 to be provided by each unit's Nuclear Steam Supply System (NSSS) sample sink and associated Post-Accident Sampling Apparatus (PASA) previously used to fulfill interim post-accident sampling capability requirements specified in NUREG-0578. By this method about 28 ml of undiluted reactor coolant is collected in a 1/2" lead shielded sample bomb at the NSSS sink (See Figure 2). The bomb is then transported to the chemistry laboratory and attached to the PASA (See Figure 3) for sample extraction. The sample then is diluted as necessary to permit chemical and isotopic analyses via normal laboratory procedures.

The Containment Atmosphere Sampling System (depicted in Figure 1) is a unshielded sample bomb assembly that is in-line with each unit's hydrogen/oxygen analyzer; and is located in the vicinity of the NSSS sink for each unit. A grab sample is expected to be taken from the septum of the bomb via syringes. The syringes are then transported to the chemistry laboratory for hydrogen/oxygen analyses via gas chromatography and isotopic analyses using normal isotopic analysis procedures.

5.3 CE-PASS System Status

The CE-PASS operability requirement is specified by Technical Specification 3/4.7.13, "Post Accident Sampling" (See Attachment 2).

Following issuance of amended Technical Specifications on February 22, 1985, the licensee declared the system inoperable on March 5, 1985, and submitted a Special Report to the NRC to that effect on March 29, 1985. This report stated that the preplanned alternate method of processing samples was in effect, and estimated that the system would be returned to operation by April 15, 1985.

On June 6, the licensee submitted another Special Report to the NRC which indicated the CE-PASS continued to be inoperable due to component failure, that the preplanned alternate sampling method was in effect, and estimated that the system would be returned to service by July 3, 1985.

(Note: At the time of this inspection, the CE-PASS was still considered inoperable due to component failure. Following this inspection, the licensee submitted another Special Report on July 22, 1985, indicating that additional problems with CE-PASS components resulted in continued inoperability, and estimated that the system would be returned to service by July 31, 1985.)

The NRC's "Order Confirming Licensee Commitments on Post-TMI Related Issues", dated March 14, 1983, documented that the CE-PASS had been taken out of service for vendor-recommended modifications and improvements; that all of the PASS sampling functions were expected to be restored by June 1, 1983; and that as an interim measure, the licensee would use the post-accident sampling system (NSSS sink/PASA method) which was in use prior to installation of the CE-PASS.

The Order further confirmed, per the licensee's submittals referenced in Section III of the Order, that the licensee would implement and maintain post-accident sampling via an upgraded post accident sampling capability (CE-PASS) by June 1, 1983.

Examination of the Facility Change Request documentation affecting the construction and establishment of the CE-PASS, and records of the systems preoperational testing failed to indicate that the system was ever verified by the licensee to be completely operational and able to perform as indicated in the submittals to the NRC. Further, these submittals were in reference only to the CE-PASS capabilities and did not infer or reference any intended use of the NSSS sink/PASA method as a backup or alternate sampling technique upon establishment of the CE-PASS as the post-accident sampling system on June 1, 1983. Such submittals were the bases of NRC's safety evaluation of the system, which concluded on February 12, 1985, that the CE-PASS system met all of the criteria

associated with NUREG-0737, Item II.B.3, "Post Accident Sampling". Subsequently, on February 22, 1985, Technical Specifications relative to the system were issued.

Based on a review of the licensee's maintenance records and discussions with cognizant personnel, it was observed that there have been continual problems with the system including leaking valves, valves which have been inoperative under system pressure, inoperable system components, and erroneous instrument indications. As a result, the system has not been available for developing and verifying procedures, and subsequent training of personnel.

No fully integrated or complete test of the system's capability to collect and analyze samples had been conducted. For example:

1. The Technical Specification 3/4.7.13 require that the system be able to collect a reactor coolant sample from the hot leg and the Low Pressure Safety Injection (LPSI); and a containment sump sample via the LPSI. Based on the data presented, only sample acquisition from the Unit 2 hot leg had been tested on June 10, 1985. At the time of this inspection documented tests to establish that samples can be collected from the LPSI at low reactor coolant pressure had not been performed.

2. The system's analytical instruments have not been tested using the NRC recommended standard test matrix solution to determine possible chemical interference that might affect analytical capability.

3. The system had not been tested to demonstrate the effectiveness of the method of sample dilution. Sample dilution capability was found on June 10, 1985 to be insufficient to permit reactor coolant chemical and isotopic analyses within the tolerances specified in licensee submittals.

4. Only one test on June 10, 1985 has been documented to establish the accuracy of the in-line analysis of radionuclides. This test revealed that just prior to the inspection, an anyalsis error of a factor of 80 for some radionuclides could be expected, and the NUREG-0737 required factor of 2 could not be generally achieved.

During the inspection, it was also noted that the germanium detector efficiencies provided by the vendor have not been verified for accuracy, and that the radiation background at post-accident conditions has not been considered in the functioning of the detector.

These items will be reviewed in a subsequent inspection (317/85-16-01; 318/85-14-01).

5.4 CE-PASS Capability Demonstration

On June 26, 1985, an exercise of the CE-PASS was attempted, but could not be performed, in that:

1. The Emergency Procedure (ERPIP 4.4.7.6) provided for the operation of the system had not been updated to agree with the current design configuration. As a result, it could not be used. Subsequently, the technicians were required to rely on the surveillance procedure (RCP-1-407) in an attempt to operate the system.

2. The licensee was unable to sample from the Unit 2 Hot Leg. It was later found that the jacking bolt to the Unit 2 Hot Leg sample acquisition valve, 2-CV-5105, had been previously tightened down on the valve in an effort to repair a chronic seat leakage problem. The valve was not tested following this "repair". The licensee's subsequent evaluation indicated that the jacking bolt had effectively locked the valve in a closed position, preventing any operation.

3. Further testing of the system was prevented when CE-PASS sample exhaust valves 1-SV-6529 and 2-SV-6529 failed to function, preventing sample flow.

These items will be reviewed in a subsequent inspection (317/85-16-02; 318/85-14-02).

5.5 Preplanned Alternate Sampling Method Status

With the operability of the post-accident sampling system, CE-PASS, less than the Limiting Condition of Operation specified in Technical Specification 3.7.13, the licensee is required to initiate the "preplanned alternate method of processing specified samples" within 72 hours.

The licensee's Special Reports to the NRC (Regional Administrator, Region I) dated March 29 and June 6, 1985 indicated that such method was in effect due to CE-PASS inoperability.

The preplanned alternate backup sampling method as described and understood from submittals to the NRC dated November 30, 1982, involved the use of the CE-PASS to provide diluted grab samples for laboratory analysis, with a provision for offsite analysis if necessary. The licensee never addressed the use of the NSSS sink with reference to the requirements of NUREG-0737 and did not indicate in submittals that the NSSS sink/PASA method would be employed as a backup to the CE-PASS. Consequently, this method was not analyzed

pursuant to the criteria of NUREG-0737, Item II.B.3; was not subject to safety evaluation, and consequently could not be considered as the "preplanned alternate method" referenced in the Technical Specifications.

However, at least since March 5, 1985, (when the licensee first declared the system inoperable pursuant to technical specification requirements) and probably earlier considering maintenance history, the NSSS sink/PASA assembly provided the licensee's only post accident sampling capability. Until July 22, 1985, it was considered by the licensee to be the "preplanned alternate method of processing specified samples" pursuant to Technical Specification 3.7.13.

The NSSS sink/PASA method was reviewed during this inspection effort to determine if the system could be considered as meeting the specification of NUREG-0737, Item II.B.3. The following was noted:

1. No approved procedure existed for the operation of the system in the present configuration.

2. No personnel have been formally trained in the NSSS sink/PASA method.

3. During a test of the NSSS sink/PASA method the operator experienced a considerable difficulty in the extraction of the sample from the bomb. Undiluted reactor coolant was forced out of the top of a burette in the PASA assembly, with a consequent loss of sample. This could have resulted in significant personnel exposure, and facility and equipment contamination in an actual post accident condition.

4. The PASA assembly was not structurally sound. The operator had to hold the assembly in place in order to apply enough force to attach the sample bomb. Additionally, all of the sample containing components of the PASA assembly, excepting tubing, were laboratory glassware. Such glassware could be subject to breakage in a post accident condition rendering the system inoperable and subjecting personnel, facilities and equipment to significant contamination.

5. The NSSS sink/PASA assembly was not provided with any shielding to reduce personnel exposure (excepting 1/2" of lead on the sample bomb). Additionally, the system has no provisions for remote operation or handling of components or processes; and required considerable direct personnel contact with components and valves that contained undiluted reactor coolant. With the exception of a shield wall and wide view mirror assembly to observe coolant flow in the NSSS sink, and a cart with lead bricks to transport the sample bomb to the radiochemistry laboratory, the method did not employ provisions to reduce personnel exposure. Though the procedure (ERPIP 4.4.7.4) did

specify the use of lead-line gloves and aprons, such equipment was not available.

6. A time and motion study to demonstrated that a sample could be collected and analyzed without exceeding GDC 19 dose criteria has not been conducted. It is unlikely that, without significant upgrading of personnel training, procedures, and ALARA considerations, the method would provide a realistic and workable option in the post accident condition.

7. The analysis procedure (ERPIP 4.4.7.4) did not contain provisions for the analysis of hydrogen and pH for post-accident samples.

8. The position indicator for the sample line isolation valve 5467 produced erratic indication. Further, the valve could not be closed after sample acquisition at the NSSS sink.

The NSSS sink/PASA method did not appear to provide a realistic and practicable option in the post accident condition; and the licensee's capability did not appear to be "preplanned" with respect to Technical Specification 3/4.7.13, in that:

1. the method may require personnel exposure beyond the dose limits of GDC 19;

2. the PASA is not structurally sound and is highly susceptible to breakage and inadvertent sample spillage;

3. the method relies upon the acceptance of high risk of radiological controls (i.e., personnel exposure, and facility and equipment contamination) and a low probability of success, particularly in successive sampling operations;

4. the method compromises the use of the NSSS sink for other sampling activities due to resultant high dose rates, since there is no sample line purge ability.

5. adequate procedure development and personnel training have not been performed.

In summary, it appears that the licensee had failed to implement and maintain the CE-PASS as required by the NRC Confirmatory Order, dated March 14, 1983; and did not implement a preplanned alternate method of sample processing sufficient for the requirements of Technical Specifications 3/4.7.13 (317/85-16-03; 318/85-14-03).

5.6 Containment Air Sampling and Analysis Capability and Status

On June 26, 1985, a containment air sample was collected. However, the following problems and concerns were noted:

1. A single common key was specified for opening two of the isolation valves. The valves could not be opened consecutively because the switches had been earlier replaced with key capture switches which prevented the use of a single key for two switches at the same time.

   Although, this change had been made about two months previous to this inspection, the information had not been incorporated in procedures or transmitted for personnel training and information.

   It was also indicated by the technician that the same set of keys should have been capable of opening valves for Unit 1 and 2; however, only one of the two valves could be opened for Unit 1 with the keys provided.

2. A technical basis for the purge times used in the procedure was not developed.

3. A flow-rate indicator has not been installed to verify gas flow in the system. A pressure indicator was relied on to indicate flow-rate. However, since pressure can exist without flow, the device did not provide positive flow indication. This was confirmed when the operator improperly aligned the valves which resulted in a pressure indication at the collection point with no flow.

4. Although remote handling tools, lead gloves and a lead-lined apron were specified for use in the procedure ERPIP 4.4.7.2, this equipment was not provided or available for use.

5. The procedure required the extraction of a gas sample from the collection bomb with a syringe that was not rated for the expected sample pressure.

6. The sample was extracted from a dead leg portion of the sample rig, which may not be representative of actual containment conditions.

7. A time and motion study was not performed sufficient to assure that the sample could be collected and analyzed within GDC-19 dose limits. The sample rig is located in the normal sampling room, which may be subject to excessive radiation levels after an accident if the NSSS sink was used for post accident coolant sampling.

8. The operator appeared unfamiliar with the control panel. This required him to expend time searching for valve switches. At one point during the test an incorrect valve was opened.

Although a containment air sample was collected, the activity was too low to permit a valid test of the licensee's analytic capability.

The following items were noted with reference to Core Damage Assessment ERPIPs:

1. Containment iodine activity is required by procedure to estimate core damage; however, the containment atmosphere sampling system design has not been evaluated to determine if a representative iodine sample can be collected via this mode.

2. The core damage procedure corrects for containment pressure and temperature. However, based on the method used to collect and analyze the sample, corrections may not be required.

This area will be examined in a subsequent inspection (317/85-16-04; 318/85-14-04).

6.0 Noble Gas Effluent Monitor, Item II.F.1-1

6.1 Position

NUREG-0737, Item II.F.1-1 requires the installation of noble gas monitors with an extended range designed to function during normal operating and accident conditions. The criteria, including the design basis range of monitors for individual release pathways, power supply, calibration and other design considerations are set forth in Table II.F.1-1 of NUREG-0737.

Documents Reviewed

The implementation, adequacy, and status of the licensee's monitoring systems were reviewed against the criteria identified in Section 4.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 1.B.

The licensee's performance relative to these criteria was determined by interviewing the principal persons associated with the design, testing, installation and surveillance of the high range gas monitoring systems, reviewing associated procedures and documentation, examining personnel qualifications and direct observation of the system design and operation.

6.2 Findings

Within the scope of this review the following were identified:

6.2.1 Description and Capability

The licensee has installed a GA Technology's, Wide Range Gas Monitor (WRGM) system for each unit to monitor noble gas releases through the main vent stacks. This system was added to compliment the pre-existing plant main vent monitor and to increase detection range. The WRGM system operationally replaces the existing noble gas detectors, however, the existing detectors are still maintained.

Separate WRGM systems are provided for monitoring the Unit 1 and 2 main vent stacks. Each system provides 3 channels of varying sensitivity to provide coverage of the desired dynamic range. The low range channel consists of an isokinetic sampling head connected by heat traced tubing to a sample conditioning module containing particulate and iodine filters. The sample then passes to the sample detection module which includes a 2 cfm pump and a plastic scintillator radiation detector. The intermediate/high range detectors have a separate sampling system sized for isokinetic sampling at 0.6 cfm, including heat traced lines and shielded iodine and particulate filters. The detectors used for this portion of the system are CdTe(Cl) directly coupled to 30 cm^3 and .02 cm^3 gas volumes.

Collectively, the detectors for the low range and intermediate/high range sample channels monitor activity concentrations from 10^{-7} to 10^5 $\mu Ci/cm^3$. They provide at least one decade of overlap between ranges.

A microprocessor is included in each WRGM system to control the sample flow rate, which filter and detector channels are used; and to compute and display release information.

A simplified diagram of the WRGM system is shown in Figure 4.

The Main Steam Effluent Radiation Monitor System (MSERM) will be used to monitor potential noble gas releases to the atmosphere from the main steam line. Two separate radiation monitors are included in the monitoring system for each Unit, providing one monitor for each steam generator. The radiation monitors are situated in the MSIV room of the Auxiliary Building to view the effluent activity in each main steam line between the steam generator and the turbine in the piping section preceeding the safety relief valves.

6.2.2 Operational Status

At the time of this inspection, the Unit 1 WRGM system was not operational due to a faulty RM-23 readout monitor in the control room. An inventory of spare parts was not available to allow

rapid replacement of the RM-23 board. The Unit 2 WRGM system was found to be operational.

The licensee has identified problems with the main vent stack flow detection instrumentation and consequently is using a default flow rate value for both WRGM systems as an input to the microprocessor for calculation of release activity. The consequences of using this default valve with respect to the system's isokinetic sampling capability requires further investigation. This will be reviewed during a subsequent inspection (317/85-16-05; 318/85-14-05).

The MSERM System had not been declared operational by the licensee at the time of this inspection. The detectors and ratemeters have been installed, but the system requires in-place calibration and the development of alarm setpoints and procedures. Additionally, review of the licensee's documentation for this system identified the following concerns:

- Calibration data showing detector response to noble gas activity rather than dose rate was not available. This data will be required to relate monitor readout (mr/hr) to main steam activity.

- Information was not available during the inspection demonstrating that the attenuation of low-energy gammas by the main steam line piping had been considered in determining monitor response.

These items will be reviewed in a subsequent inspection (317/85-15-06; 318/85-14-06).

6.2.3 Procedures and Training

Review of licensee procedures and associated training involving the WRGM system indicated that procedures and training are in an early stage of development. Specifically, the following was identified.

- The majority of necessary Surveillance Test Procedures and associated Preventive Maintenance procedures for the WRGM system have not been developed.

- The licensee's Emergency procedures do not specifically reference the use of the WRGM system as an input method for obtaining stack release rate. Additionally, a study evaluating WRGM detector response to varying isotope mixes predicted for varying time intervals after the accident was not evaluated by Emergency Planning as to effect on offsite dose calculations.

- The individual displaying WRGM control room readouts during this inspection could not interrogate the system.

Additionally, no formalized procedures or training had been developed to control activities associated with the operation or maintenance of the MSERM and WRGM systems. Licensee development of procedures and training in this area will be reviewed in a subsequent inspection (317/85-16-07; 318/85-14-07).

6.3 Acceptability

The licensee's WRGM system provisionally fulfills the noble gas monitoring requirements of NUREG-0737, II.F.1-1, pending correction of the discrepancies identified above. Due to the non-operational status of the MSERM System, the licensee has yet to demonstrate capability for monitoring noble gas releases through the steam line.

7.0 Sampling and Analyses of Plant Effluents, Item II.F.1-2

7.1 Position

NUREG-0737, Item II.F.1-2, requires the provision of a capability for the collection, transport, and measurement of representative samples of radioactive iodines and particulates that may accompany gaseous effluents following an accident. Such activities must be performable without exceeding specified dose limits to the individuals involved.

The criteria including the design basis shielding envelope, sampling media, sampling considerations, and analysis considerations are set forth in NUREG-0737, Table II.F.1-2.

Documents Reviewed

The implementation, adequacy and status of the licensee's sampling and analysis system and procedures were reviewed against the criteria identified in Section 4.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 1.B.

The licensee's performance relative to these criteria was determined by interviewing the principal persons associated with the design, testing, installation, and surveillance of the systems for sampling and analysis of high activity radioiodine and particulate effluents, by reviewing associated procedures and documentation, by examining personnel qualifications, and by direct observation of the system design and operation.

7.2 Findings

Within the scope of this review, the following items were identified:

7.2.1 Descriptions and Capability

The licensee is intending to use the grab-sample capability of the Wide Range Gas Monitoring (WRGM) system to meet the radioiodine and particulate sampling requirements of NUREG-0737, Item II.F.1-2. As described previously in section 6.0, the licensee maintains two independent WRGM systems for independent monitoring of each unit's main vent stack. Each system samples the main vent stack with two separate sampling pathways; a high flow, low range pathway and a low flow, intermediate/high range pathway.

Both the low range and the intermediate/high range sample pathways of the WRGM system feature a grab-sample capability with quick disconnect particulate and iodine (silver zeolite) filters. Filters on the intermediate/high range pathway are enclosed in shielded casks which can be removed from the sampling skid for transport. Diversion of the effluent sample through the grab-sample filters and timing of the grab-sample duration can be controlled remotely from the control room.

The high flow (2 scfm) low range sampling pathway line for each WRGM system is made of 3/4 inch outer diameter stainless steel. The low flow (0.06 scfm), intermediate/high range sampling pathway is made of 1/4 inch outer diameter stainless steel. These lines are nearly 200 feet in length and contain about a dozen right angle bends. The sampling lines are heat traced from each stack to the sampling filter skids.

In light of the sampling line lengths and pathways, as described, significant radioiodine and particulate line losses due to plateout and deposition can be expected for this sampling configuration. However, the licensee has not performed a line loss evaluation to quantify the loss and determine appropriate correction factors. Currently, the licensee assumes 100% transmittal efficiency through the sampling lines. This item will be reviewed in a subsequent inspection (317/85-16-08; 318/85-14-08).

An operability surveillance requirement for the main vent iodine and particulate sampling capability was established as an amendment to the Technical Specifications on February 22, 1985. Technical Specification Section 4.3.3.8 requires that, "the main vent iodine and particulate sampler shall be demonstrated operable by comparing samples independently drawn from the main vent at least once per month." The inspector determined by discussion with licensee chemistry personnel that the required

comparison has never been performed to demonstrate operability of either the Unit 1 or Unit 2 sampling capability. This finding appears to constitute a violation of NRC requirements (317/85-16-9; 318/85-14-9).

The licensee is currently relying on the installed heat tracing on the sampling lines to prevent condensation and subsequent sampling problems with the sample lines. However, no evaluation has been performed to demonstrate the capability of the heat tracing to provide adequate heating to the sample lines under all ambient temperature conditions. Additionally, at the time of this inspection the heat tracing for the Unit 1 WRGM system was found to be non-operational. The licensee indicated by subsequent telephone conversation on July 29, 1985 that this had been corrected.

7.2.2 Capability Demonstration

A demonstration of grab sample acquisition, retrieval, and analysis was performed during this inspection using procedure RCP 1-405. Two chemistry technicians were assigned the task of retrieval of the grab sample filter casks. The casks were then transported to the chemistry laboratory for analysis. The inspector identified the following problems during the filter retrieval and analysis operation.

- The technicians had not received formal training in the procedure describing grab sample retrieval.
- The procedure did not specify tools or equipment that might be required. One technician had to exit the area during the operation to obtain a wrench. This may cause additional personnel exposure during accident conditions.
- The procedure did not provide guidance relative to personnel exposure considerations.
- Filter cask removal from the skid was difficult to perform and required two technicians approximately 25 minutes. During this period, the technicians were in close proximity to the grab sample cask and other scrubbing filters on the sampling skid.
- No procedures were available to control handling of the shield assembly or analysis of the filters in the laboratory.
- No remote manipulating tools were available in the chemistry laboratory to handle the filters.

A letter from D. G. Weiss (General Atomics) to H. B. Wylie (BG&E), dated 4/26/82, indicated exposure from retrieval of the

grab sample cask would be less than 5 rem whole body. Weiss assumed work at a distance of one meter from a single cask. The sample was 30 minutes in duration and activity levels were assumed to be 2×10^2 uCi/cm^3. The flow rate was 0.06 scfm. The Weiss study did not account for 1) the time and motion involved in separating the cask from the skid, 2) the proximity of the person to the surface of three casks on the skid, and 3) the activity of radioiodine on the scrubbing filters.

Concerns listed above, that were identified during the licensee's demonstration of their grab sample capability will be reviewed in a subsequent inspection (317/85-16-10; 318/85-14-10).

Currently, the grab sample capability of the WRGM system is less than adequate in fulfilling the requirements of NUREG-0737, Item II.F.1-2. This system had not been demonstrated to provide representative samples, surveillance requirements have not been implemented, procedures and personnel training are insufficient, and personnel exposure considerations are not complete.

8.0 Containment High Range Radiation Monitor, Item II.F.1-3

8.1 Position

NUREG-0737, Item II.F.1-3 requires the installation of high range radiation monitors capable of detecting and measuring radiation levels within the reactor containment during and following an accident. Specific requirements are set forth in NUREG-0737, Table II.F.1, Attachment 3.

Documents Reviewed

The implementation, adequacy and status of the licensee's containment high range radiation monitoring system was reviewed against criteria identified in Section 4.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 1.C.

8.2 Description

The licensee has installed two General Atomics, Incorporated (GA), Model RD-23 Gamma Radiation Detectors (gamma ionization chambers) in each reactor unit. Model RP-2C Readout Modules are installed in each unit's control room. The RD-23 is capable of detecting radiation from 10^0 to 10^8 R/hr. As a safety monitor, it satisfied Class 1E requirements and is qualified under LOCA conditions per IEEE 323-1974. The detectors are encased in stainless steel to protect them from containment sprays and high temperatures.

The following specifications apply:

Parameter	Description
Range	10^0 to 10^8 R/hr
Sensitivity	~ 1×10^{-11} amp/R/hr
Max. Temperature	350°F
Max. Pressure	70 psig
Humidity	0-100% (saturated steam)
Seismic Qualification	Per IEEE 344-1975

Adequate vendor calibration sufficient per instrument type certification was verified. Though the system is not yet subject to technical specifications, the licensee conducts monthly surveillance tests to verify operability, i.e., STP-O-98-1[2], "Containment High Range Monitor Monthly Test." Vital Class IE power sources are used for each instrument channel.

Startup and operation of the system is described in OI-35, "Radiation Monitoring System", which is supported by formalized lessons plans for personnel training. Calibration test procedures are performed at refueling outages in accord with M-562-1, "Containment High Range Monitor Alignment Check" and M-563-1, "Containment High Range Monitor Source Check." The source check procedure subjects the instrument to two dose rates within the 10 R/hr range.

Two independent monitors are located in each containment at the 73' elevation. The monitors are on No. 12 & 22 Steam Generator cubicles, and on each unit's Pressurizer cubicle. Sufficient view and volume appears to be monitored by this configuration.

8.3 Findings

Submittals made to the NRC with respect to this monitoring requirement and incorporated in the NRC Confirmatory Order dated March 14, 1983, indicated that the installation was as prescribed by NUREG-0737, i.e., developed and qualified to function in an accident environment. To this end, each instrument channel is expected to be comprised of an instrument assembly (i.e., detector, detector-to-cable connector, cable, and cable-to-penetration connector) that is environmentally qualified for LOCA conditions.

On June 26 and 27, 1985, it was verified that the cable-to--penetration connectors for each channel in Unit 1 were not sleeved with any RAYCHEM sleeving material. Such sleeving was necessary to assure that the Amphenol cable connector (part #82-816), which was used in lieu of the originally specified part identified in Field Change Request (FCR) 79-1057, would be environmentally qualified. The Field Equipment Change (FEC) 79-1057-6(p) failed to specify the sleeving, though required for environmental qualification. The result of this oversight is that Unit 1 installation was not

sufficient to assure the capability of detecting and measuring radiation levels within the Unit 1 reactor containment during and following an accident, for the period between March 14, 1983 (the date of the NRC Confirmatory Order), and June 26, 1985.

The licensee later confirmed to NRC management that the proper sleeving was originally installed properly on the Unit 2 cable-to-penetration connectors; and the Unit 1 deficiency had been corrected. Other aspects of this area appear to be in accord with the requirements specified in NUREG-0737.

This item appears to constitute a violation of NRC requirements (317/85-16-11; 318/85-14-11).

9.0 III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions

9.1 Position

NUREG-0737, Item III.D.3.3 requires that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Technical Specifications 6.15, "Iodine Monitoring" requires the licensee to implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program requires the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

The implementation, adequacy, and status of the licensee's in-plant iodine monitoring under accident conditions was reviewed against the criteria in Section 4.0 and in regard to the following documents:

-- Procedure No. ERPIP 4.1.5, Revision 9, "Radiation Protection Director."

-- Procedure No. ERPIP 4.1.6, Revision 11, "Offsite Monitoring Team".

-- Procedure No. ERPIP 4.1.7, Revision 9, "Onsite Monitoring Team".

-- Training Instruction 5, "Emergency Response Training Program".

-- Lesson Plan No. ER-2-6, "SPEC PRO".

9.2 Findings

An onsite monitoring team is designated to perform in-plant radiological surveys, including radioiodine monitoring, in post-accident conditions. Twelve technicians have been selected to perform the duty as part of the licensee's program for emergency response.

A portion of the licensee's program to ensure the capability to accurately determine iodine concentration in post-accident conditions is defined by Training Instruction 5, which requires yearly training for personnel assigned to the onsite monitoring team.

From review of personnel training records it was noted that the last training to be performed in this area was conducted during February, 1984; and at that time only seven of twelve designated personnel were provided with such training. This appears to be contrary to the licensee's program requirements of Technical Specification 6.15, and associated implementing procedures (317/85-16-12/ 318/85-14-12).

The licensee indicated that ERPIP 4.1.7 defines the scope of activities for the Onsite Monitoring Team. Review of this procedure revealed that 1) in-plant radioiodine monitoring was not addressed in the procedure, and 2) the procedure did not detail any in-plant monitoring activities. This item will be reviewed in a subsequent inspection (317/85-16-13; 318/85-14-13).

Relative to other areas affecting in-plant radioiodine monitoring, it was noted that the licensee had provisions for the maintenance of sampling and analysis equipment. Such equipment appeared to be of sufficient type, quality and quantity to assure an adequate capability.

Silver zeolite cartridges are used for iodine sampling during post-accident conditions, and are subject to clean air purging prior to radioanalysis.

10.0 Exit Interview

The inspector met with the licensee management representatives (denoted in Section 1.0) at the conclusion of this inspection on June 28, 1985, to discuss the scope and findings of the inspection as detailed in this report. At that time, the licensee's representatives were informed that a management meeting to discuss these findings and the licensee's corrective actions would be held on July 11, 1985, at NRC Region I.

Attachment I.A.

Documentation for NUREG-0737, II.B.3

Correspondence

-- A. E. Lundvall, Jr., VP BG&E, to Robert A. Clark, Chief, ORB #3, DOL, dated May 21, 1980.

-- A. E. Lundvall, Jr., VP BG&E, to D. G. Eisenhut, Dir. DOL, dated December 15, 1980.

-- A. E. Lundvall, Jr., VP BG&E, to D. G. Eisenhut, Dir. DOL, dated July 7, 1981.

-- A. E. Lundvall, Jr., VP BG&E, to D. G. Eisenhut, Dir., DOL, dated November 23, 1981.

-- R. A. Clark, Chief ORB #3, DOL to A. E. Lundvall, Jr. VP BG&E, dated March 15, 1982.

-- A. E. Lundvall, Jr., VP BG&E, to D. G. Eisenhut, Dir. DOL, dated March 26, 1982.

-- A. E. Lundvall, Jr., VP BG&E, to D. G. Eisenhut, Dir. DOL, dated April 19, 1982.

-- R. A. Clark, Chief ORB #3, DOL, to A. E. Lundvall, Jr. VP BG&E, dated June 30, 1982.

-- A. E. Lundvall, Jr., VP BG&E, to R. A. Clark, Chief ORB #3, DOL, dated August 6, 1982.

-- A. E. Lundvall, Jr., VP BG&E, to R. A. Clark, Chief ORB #3, DOL, dated November 30, 1982.

-- R. W. Starostecki, NR[illegible], to A. E. Lundvall, Jr., VP BG&E, dated June 25, 1984.

-- A. E. Lundvall, [illegible] BG&E, to J. R. Miller, ORB #3, DOL, dated June 29, 1984.

-- A. E. Lundvall, Jr., VP BG&E, to J. R. Miller, ORB #3, DOL, dated October 30, 1984.

-- M. D. Patterson, BG&E to J. R. Miller, ORB #3, dated January 15, 1985.

-- J. R. Miller, ORB #3 to A. E. Lundvall, Jr., VP BG&E, dated February 12, 1985.

-- D. H. Jahl, ORB #3 to A. E. Lundvall, Jr., VP BG&E, dated February 22, 1985.

NRC Memoranda

-- W. V. Johnson, NRRET to G. C. Lainas, NRRLO, dated June 24, 1983.

-- D. M. Crutchfield NRRLS to G. C. Lainas, NRRLO, dated November 6, 1984.

-- W. V. Johnson, NRRET to G. C. Lainas, NRRLO, dated January 17, 1985.

Procedures

-- RCP 2-102, "Operation of the Nuclear Data 6620 System", Rev. 10, dated January 6, 1984.

-- RCP 2-103, "Operation of the Tracor Northern-11 System", Rev. 10, dated November 21, 1984.

-- RCP 2-105, "Operation of P.A.S.S. Gamma Detector System", Rev. 10, dated June 20, 1985.

-- RCP 1-407, "Post Accident Sampling System Operation & Analysis", Rev. 10, dated May 29, 1985.

-- TSP 104, "Flush, Hydrostatic, Pneumatic and Functional Testing of Modifications to the PASS", Rev. 0, dated June 29, 1983.

-- TSP 64, "PASS Flush, Hydro, and Valve Verification", Rev. 1, dated May 17, 1982.

-- TSP 58, "Post Accident Sampling System Preliminary Testing", Rev. 0, dated August 11, 1982.

-- TSP 69, "Post Accident Sampling System (PASS) Preoperational Testing", Rev. 0, dated May 28, 1982.

-- ERPIP NO.: 4.4.7.3, "Post Accident Reactor Coolant Sampling", Rev. 9.

-- ERPIP NO.: 4.4.7.6, "Post Accident Sampling System and Analysis, Rev. 2.

-- ERPIP NO.: 4.4.7.1, "Containment RMS Reading Versus Time Following Accidents", Rev. 9.

-- ERPIP NO.: 4.4.7.2, "Post-Accident Containment Atmosphere Sampling", Rev. 10.

-- ERPIP NO.: 4.4.7.3, "Post-Accident Reactor Coolant Sampling, Rev. 10.

-- ERPIP NO.: 4.4.7.4, "Post-Accident Reactor Coolant Analysis, Rev. 9

-- ERPIP NO.: 4.4.7.5, "Post-Accident Hydrogen Analysis", Rev. 1.

-- ERPIP NO.: 4.4.5, "Initial Determination of Projected Whole Body Doses, Rev. 8.

-- ERPIP NO.: 4.1.3.1, "Core Damage Assessment", Rev. 8.

-- ERPIP NO.: 4.1.3.2, "Core Damage Assessment", Rev. 8.

-- ERPIP NO.: 4.1.3.3, "Core Damage Assessment", Rev. 8.

-- ERPIP NO.: 4.1.3.4, "Core Damage Assessment", Rev. 8.

-- ERPIP NO.: 4.1.3.5, "Core Damage Assessment", Rev. 8.

Drawings

-- M-66, "Reactor, Coolant & Waste System Units 1 & 2", Sheets 1, 2 & 3 Rev. 15, dated October 23, 1984.

-- M-66, "M-72, Reactor and Coolant System Unit No. 1", Rev. 17, dated January 23, 1985.

-- M-74, "Safety Injection and Containment Spray Systems", Sheets 1, 2, & 3, Rev. 36, dated December 10, 1984.

-- M-77, "Reactor, Coolant & Waste System Units 1 & 2", Sheets 1, 2, & 3, Rev. 20, dated September 6, 1984.

-- M-463, "Gas Analyzing System Units 1 & 2", Sheets 1 & 2, Rev. 14, dated October 15, 1984.

Attachment I.B.

Documentation for NUREG-0737, II.F.1-1,2

Calvert Cliffs Nuclear Power Plant Emergency Response Plan

-- ERP Pages 5-3, 5-4, 5-5, 5-6, 5-7, 5-8.

Calvert Cliffs Nuclear Power Plant Emergency Implementation Procedures

-- ERPIP 3.0, "Radioactivity Release Quick Estimate", Rev. 11.

-- ERPIP 4.4.6, "Initial Estimate of Fission Product Release Based on Environmental Measurements", Rev. 2, dated September 1, 1981.

-- ERPIP 4.4.3, "Initial Determination of Accident Radioactivity Release", Rev. 8, dated October 28, 1981.

Calvert Cliffs Nuclear Power Plant Final Safety Analysis Report

-- 11.2.3, "Radiation Monitoring", Rev. 2.

Calvert Cliffs Nuclear Power Plant Operating Procedures

-- RCP 1-405, "Operation of the Main Vent Wide Range Noble Gas Monitors", Rev. 0.

Other Licensee Documents

-- OI-48, "Noble Gas Monitor"

-- BG&E drawing #60-275-E (Bechtel #M-98-Sh)

Vendor Manuals

-- "General Atomics Calibration Report RD-72 Wide-Range Gas Monitor High and Mid-Range Detectors", E-255-961 (Rev. 2).

-- General Atomics WRGM Equipment Manual

Licensee Correspondence

-- A. Lundvall, Jr., VP BG&E, to R. Clark, Chief ORB, NRC, dated April 13, 1982.

-- D. G. Weiss, General Atomics to B. Wylie, BG&E, dated April 26, 1982.

-- A. Lundvall, Jr., VP BG&E to R. Clark, Chief ORB NRC, dated February 18, 1983.

-- A. Lundvall, Jr., VP BG&E to J. Miller, Chief ORB NRC, dated February 3, 1984.

-- A. Lundvall, Jr., VP BG&E to J. Miller, Chief ORB NRC, dated February 16, 1984.

-- A. Lundvall, Jr., VP BG&E to J. Miller, Chief ORB NRC, dated November 1, 1984.

-- L. Russell, Plant Superintendent BG&E to T. Murley, Regional Administrator, NRC, dated April 4, 1985.

-- A. Lundvall, Jr., VP BG&E to J. Miller, Chief ORB NRC, dated May 9, 1985.

NRC Correspondence

-- R. Clark, Chief ORB NRC to A. Lundvall, VP BG&E, dated September 30, 1981.

-- R. Clark, Chief ORB NRC to A. Lundvall, VP BG&E, dated November 2, 1981.

Attachment 1.C

Documentation for NUREG-0737, II.F.1-3

E-115-876, "High Range Gamma Radiation Monitoring System Operation and Maintenance Manual," dated June 1981, Revised October 1, 1984

OI-35, "Radiation Monitoring System" Revision 7, dated May 24, 1985

STP-0-98-1[2], "Containment High Range Monitor Monthly Test", Revision 1, dated May 1, 1985

M-562-1, "Containment High Range Monitor Alignment Check"

M-563-2, "Containment High Range Monitor Source Check"

Field Change Request 79-1057, "Containment High Range Radiation Monitors"

PLANT SYSTEMS

3/4.7.13 POST-ACCIDENT SAMPLING

LIMITING CONDITION FOR OPERATION

3.7.13 The post-accident sampling system shall be OPERABLE and capable of processing samples from all of the below listed points:

a. RCS sample via hot leg

b. RCS sample via low pressure safety injection, and

c. Containment sump sample via low pressure safety injection.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the operability of the post-accident sampling system less than the LIMITING CONDITION FOR OPERATION specified above, within 72 hours initiate the preplanned alternate method of processing specified sample(s), and either:

1. Restore the system to OPERABLE status within 7 days, or

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.13 The post-accident sampling system shall be demonstrated OPERABLE at least once per six (6) months by comparing the results of a RCS sample analyzed by laboratory techniques with the results analyzed by the below listed analyzing equipment:

1. Boron Analyzer

2. Hydrogen and Oxygen Analyzer

3. pH Analyzer

4. Liquid Radioisotopic Analyzer.

FIGURE 1

COMBUSTION ENGINEERING POST ACCIDENT
SAMPLING SYSTEM

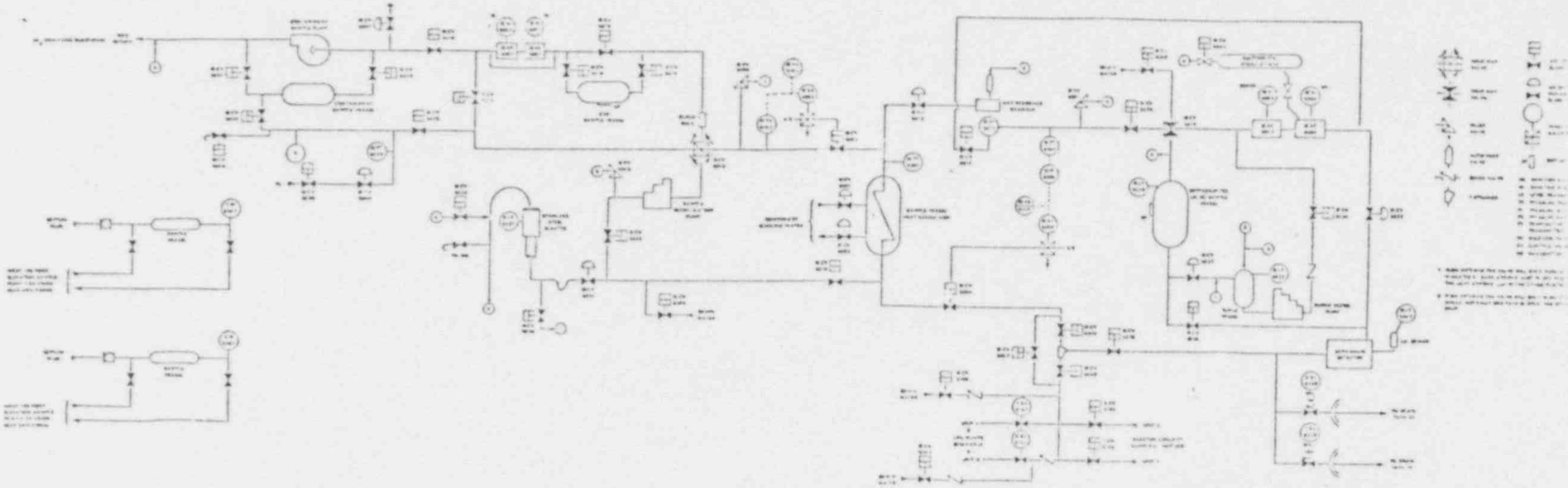


FIGURE 2

POST ACCIDENT RCS COOLANT

SHIELDED COOLANT SAMPLE COLLECTION APPARATUS

(SAMPLE BOMB)

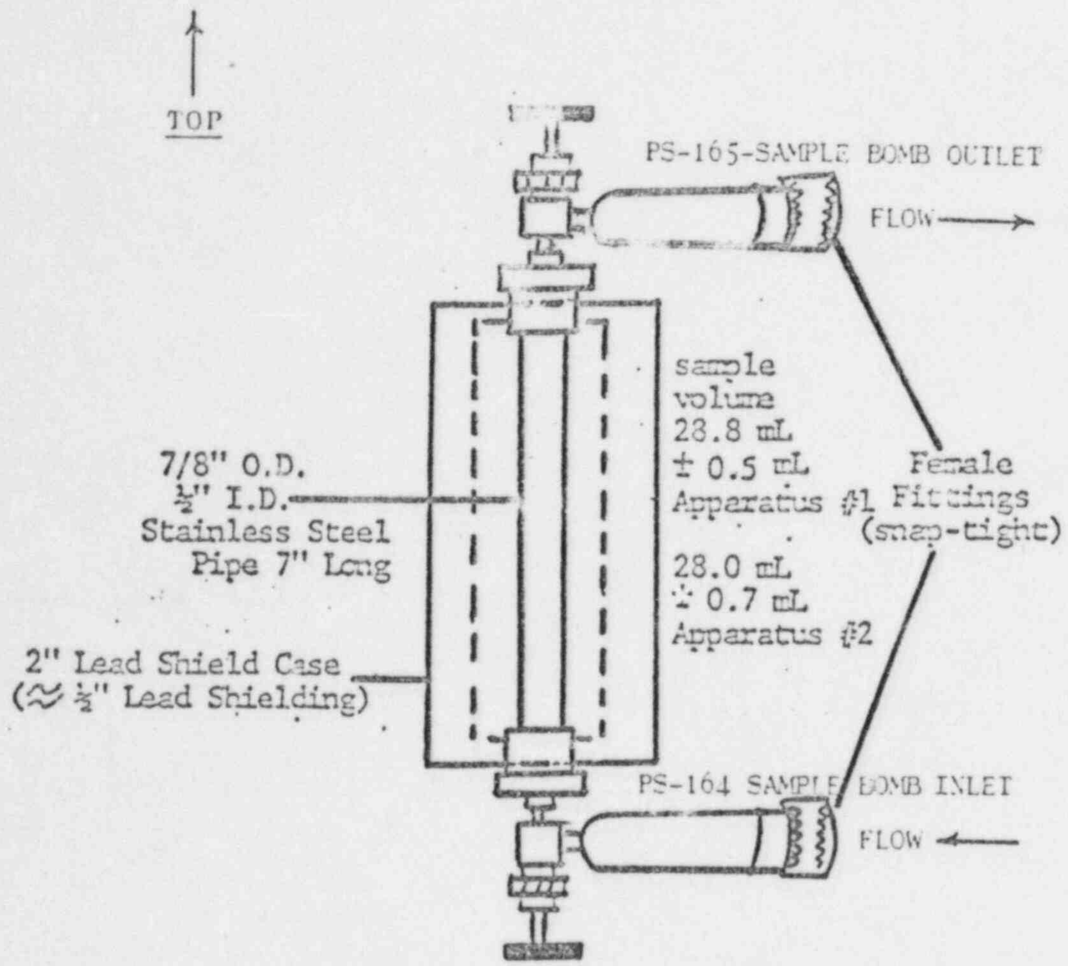
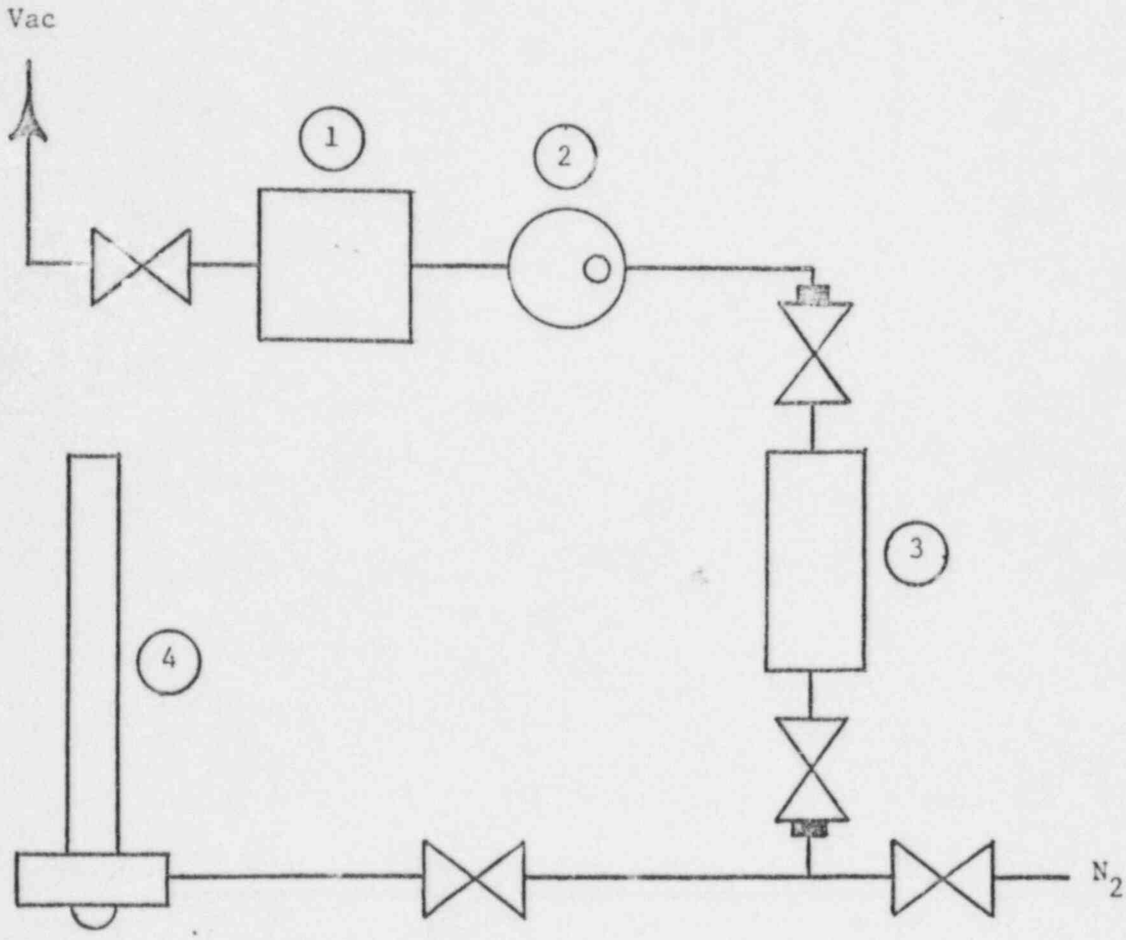


FIGURE 3

"POST ACCIDENT SAMPLING APPARATUS"



1. Charcoal Cartridge
2. Glass Gas Bulb
3. Shielded Post Accident Sample Collection Apparatus
4. B irette

FIGURE 4
WIDE RANGE GAS MONITORING SYSTEM