U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-219/86-01

Docket No. 50-219

License No. DPR-11 Category C

Licensee: GPU Nuclear Corporation
P.O. Box 388
Forked River, NJ 08731

Facility Name: Oyster Creek Nuclear Station

Inspection At: Forked River, New Jersey

Inspection Conducted: January 13-17, 1986

Inspectors: Marie Miller — 2/6/86
M. Miller, Radiation Specialist — date

R. Sherbini for — 2/19/86
S. Sherbini, Radiation Specialist — date

R. Paolino — 2-6-86
R. Paolino, Lead Reactor Engineer — date

M. Miller for — 2/6/86
A. P. Hull, Brookhaven National Laboratory — date

M. Miller for — 2/6/86
S. V. Musolino, Brookhaven National Laboratory — date

Approved by: M. Shanbaky — 2/11/86
M. Shanbaky, Chief, Facilities Radiation Protection Section — date

Inspection Summary: Inspection on January 13-17, 1986 (Report No. 50-219/86-01)

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 201 hours by three region-based inspectors and two contractors from Brookhaven National Laboratory.

Results: No violations were identified. Several areas requiring improvements and further review were identified.

DETAILS

1.0 Persons Contacted

1.1 General Public Utilities

J. Anderscavage,	Scheduling Supervisor
*W. Behrle,	Director, Start-up and Test
J. Bishop,	Start-up Engineer
*G. W. Busch,	Licensing Engineer
M. Buday,	Manger Plans & Programs
J. Carscadder,	Consulting Engineer
D. Chandler,	Engineering Process and Instrumentation
*W. Duda,	Projects Manager
W. Dunphy,	Senior Chemist
*S. C. Gera,	Project Engineer
*P. B. Fiedler,	Vice President/Director
*C. J. Halbfoster,	Manager, Plant Chemistry
R. Hillman,	Senior Chemist
*B. Hohman,	Licensing Engineer
T. Johnson,	Area Supervisor - Electrical
*R. W. Keaton,	Director Engineering Projects
A. Lewis,	Document Control Supervisor
M. Littleton,	Manager, Radiological Engineering
R. Parshall,	Administrative Support Supervisor
*M. J. Radvansky,	Manager, Technical Functions
*G. J. Sadauskas,	Manager, Instrumentation & Controls
*G. J. Simonetti,	Audit Manager
*J. Solakiewicz,	Manger, Quality Assurance and Systems
*J. Stevens,	Process Instrumentation
R. Stoudnour,	Senior Engineer
*J. L. Sullivan Jr.,	Plant Operations Director
*R. L. Sullivan,	Manager, Emergency Preparedness
*J. Thorpe,	Director, Licensing and Regulatory Affairs
*D. Turner,	Radiation Control Director
M. Wineberg,	Technical Functions Engineer

1.2 Nuclear Regulatory Commission

W. Pasciak,	Chief, Effluents Radiation Protection Section
B. Bateman,	Senior Resident Inspector, OC
J. Wechselberger,	Resident Inspector, OC

*denotes attendance at exit interview on January 17, 1986.

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 Plant 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

As part of the inspection, a review was performed to verify and validate the adequacy of the licensee's design and quality assurance program for the design and installation of the Post-Accident Sampling System (PASS).

3.0 TMI Action Plan Generic Criteria and Commitments

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

- NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, dated July 1979.
- Letter from Darrell G. Eisenhut, Acting Director, Division of Operating Reactors, NRC, 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 Darrell G. Eisenhut, Director, Power Reactors, dated March 14, 1982.
- Letter from Darrel G. Eisenhut, Director, Division of Licensing, NRR to Regional Administrators "Proposed Guidelines for Calibration and Surveillance Requirements for Equipment Provided to Meet Item II.F.1, Attachments 1, 2 and 3, NUREG-0737" dated August 16, 1982.
- Order confirming Licensee Commitments on Post-TMI Related Issues, dated June 17, 1983.
- Oyster Creek Nuclear Generating Station, Updated Final Safety Analysis Report, dated December 1984.
- Modifications of Confirmatory Order of June 17, 1983 for II.B.3, Post-Accident Sampling System dated April 29, 1985.
- Regulatory Guide 1.3 "Assumptions Used for Evaluating Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.

- Regulatory Guide 1.97 Rev. 3, "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 Stations will be As Low As Reasonably Achievable".

4.0 Post-Accident Sampling System, Item II.B.3.

4.1 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 3.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 1 of this Inspection Report.

The licensee's performance relative to these criteria was determined from interviews with the principal personnel associated with post-accident sampling, reviews of associated procedures and documentation, and the conduct of a performance test to verify hardware, procedures and personnel capabilities.

4.2 Findings

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

4.2.1 System Description and Capability

The licensee has installed a Post-Accident Sampling System which is a standard General Electric design. It has the capability to obtain undiluted and diluted unpressurized liquid samples. They may be drawn from the reactor vessel through the regular and through the liquid poison sampling lines, from the shut-down heat exchange system and from the torus via the core-spray system. Atmosphere samples can be obtained from the drywell, suppression pool and reactor building (secondary containment). The PASS sampling cabinet and control panel are situated in a room just outboard of the reactor building.

Analysis for radioactivity is conducted in an adjacent laboratory using a Canberra Series 85 high resolution system

with a Ge-(Li) detector and computerized MCA system. Analysis for chlorides, boron and hydrogen are also conducted in an adjacent laboratory, using an ion-chromatographic method, the carminic acid method and gas chromatography, respectively. The PASS also includes a capability for on-line conductivity measurement. Analysis for pH is conducted using micro-electrode and a small aliquot of a 10-ml undiluted sample.

The licensee was originally committed in the Confirmatory Order dated June 17, 1983, to having the PASS operational within 6 months after startup from the Cycle 10 refueling outage. Subsequently, the licensee discovered leakage of a valve (40-29) in the recirculation system sampling line, which required isolation in accordance with Technical Specifications, so was unable to fully test the system at that time.

A modification of this Confirmatory Order was made on April 29, 1985 to extend the date to no later than the planned shutdown for October 1985. The valve in question had been repaired and the licensee completed operational testing on November 18, 1985.

Reactor coolant and drywell sampling have been conducted in which samples from the PASS have been compared with these from the normal sampling locations. Flow tests through other sample lines were not compared due to low levels of radioactivity. However, all sample pathways were tested by physical techniques (i.e. flow of demineralized water or freon injected under pressure at special test taps in sample lines).

4.2.2 Performance Test

Grab samples of reactor coolant and of the drywell atmosphere were obtained in a performance test for this inspection on January 15, 1986. During the test, licensee personnel verified the integrated ability to collect and analyze samples within the time constraints of NUREG-0737, II.B.3.

4.2.3 Sampling

4.2.3.1 Reactor Coolant

The reactor coolant sampling system is designed to obtain samples of liquids and dissolved gases during all modes of operation. The following findings were noted:

- The volume actually delivered by the ball valve in the small-sample dilution procedure, which is specified by the vendor to be 0.1 ml, has not been verified by the licensee.

- The procedure for the drawing of a sample of dissolved gas (8.10, Appendix 3, Step 3.55.3) calls for the operator to "grab the knurled portion of the needle when removing the syringe". It does not contain a precaution against contact with the portion of the needle which may become contaminated during the test.

- Guide marks have been improvised in pencil on the wall below and behind the PASS sampling cabinet to guide the positioning of the large cart bearing the shield which is utilized for the undiluted sample procedure (8.10, Appendix 2 and Appendix 3).

- Although the indications of the radiation monitor for liquid samples (RI-665) are utilized in the procedures for sampling (8.10, Appendix 2 and 3) to assure that flushing has taken place, it is not specifically referred to as an indication for the operator that a high activity sample has been collected.

4.2.3.2 Containment Air

Atmosphere samples can be obtained from the Drywell, Reactor Building and the Torus. The following findings were noted:

- The licensee procedure for taking an iodine and particulate sample (831.19, Appendix 5) calls for the determination of the flow by means of the reading of a rotometer (FI-725, which is incorrectly designated in the procedure as PI-175 and which also incorrectly calls for a flow reading of GPM, instead of SCFM). However, in correspondence to another utility (T.A. Green, Manager, Servicing and Auxiliary Equipment Retrofits GF to T.H. Wyllie, Manager Brunswick Engineering CP&L, dated April 6, 1984), it is stated that this rotometer is used "strictly to verify gas purge flow as the critical flow orifice is used as the accurate flow measurement device during particulate and iodine sampling".

- The indications of the radiation monitor (RI-704 for the particulate/iodine cartridge) are not specifically called out in the procedure to alert the operator that a high activity sample has been collected.

- The iodine sampling cartridge depends on a metal to metal contact of four individual in-line iodine filter canisters under modest compression to prevent streaming past them.

4.2.3.3 Recommendations for Improvement

A. Verify the volumetric delivery of the ball valve for a diluted sample.

B. An appropriate caution against contact with the needle should be added to the procedure for the drawing of a sample of dissolved gas.

C. Clearly visible guidance should be provided and described in the procedure for the positioning of the large cart for undiluted samples.

D. The indications of the radiation detectors installed adjacent to the liquid sample and the particulate/iodine sampling cartridge should be utilized to alert the operator to the collection of high activity samples.

E. The procedures for the determination of the flow in collection of particulate/iodine samples should be based on the flow through the critical orifice, with an appropriate precaution that the appropriate pressure differential (approximately 0.5 cfm) is observed.

F. The use of O rings between the canisters in the iodine sampling cartridge should be considered, unless it can otherwise be demonstrated that by-pass leakage cannot occur.

This item will be reviewed during a subsequent inspection (219/86-01-01).

4.2.4 Analytical Capability

The licensee's commitment relative to range, sensitivity and type of analytical capability as indicated in Appendix B, were contained in its submittals of March 6 and July 13, 1984.

4.2.4.1 Chloride

Preliminary screening for chlorides is performed using ion chromatography. In the event of interference due to a high ratio of boron to chloride, a separation would be performed and the turbidimetric method utilized. Backup off-site analysis capability would be available through an agreement with B&W's Lynchburg, VA Laboratory. A shipping cask is available. However, a certificate of conformance relating to the quality assurance program for the cask was not documented.

Chloride analysis was satisfactorily conducted with the ion chromatography method. The results are contained in Attachment 2. However, the license stated in Procedure 824.9, "Chemical Instrumentation: Ion Chromatography" that the lower limit for detection was 0.1 ppb. This value could not be demonstrated and apparently was a typographical error.

4.2.4.2 Boron

Boron analysis of PASS samples is performed using the carminic acid method. Mannitol titration would be used for low concentration samples.

Boron analyses was satisfactorily conducted with the carminic acid method. The results are contained in Attachment 2.

4.2.4.2 pH Analysis

Analysis for pH is conducted in a hood adjacent to the PASS sampling unit using a micro-electrode that can utilize samples as small as 0.1-0.3 ml. The licensee demonstrated the capability of the micro-electrode using 0.1 ml sample size. The results are contained in Attachment 2.

4.3.3.4 Gross Gamma and Isotopic Analysis

Gamma analysis of PASS samples is performed using a Canberra Series 85, computer based, high-resolution system with a shielded Ge-(Li) detector. An extensive library is utilized which is sufficient to detect the nuclides of interest. By the use of dilution and small shielded sample transport containers with bottom apertures and an adapter on the detector shield, the full range of anticipated concentrations can be evaluated.

An isotopic analysis of the undiluted reactor coolant sample was satisfactorily conducted. The results of the comparison of the PASS sample and the normal sink sample compared within a factor of two. The licensee had not completed its site specific core damage estimate procedure. However, a methodology based on the GE core damage estimate procedure was available for iterim use.

4.2.4.6 Hydrogen and Dissolved Gas

Dissolved gas is determined by the GE PASS expansion method. Hydrogen and/or oxygen content are evaluated by gas chromatography.

The licensee satisfactorily demonstrated the ability to collect a dissolved gas sample and to perform hydrogen analysis with gas chromatography. The results are contained in Attachment 2.

4.2.5 Additional Findings

A. Calibration and Maintenance

According to the licensee submittal of March 6, 1984 (Criterion 10, Item 7), "Equipment used for post-accident sampling and analysis will be calibrated or tested approximately every six months". However, it could not be verified that a schedule for calibration or testing had in fact been established. It was noted by the inspector that several instruments on the PASS panel had calibration stickers with dates a year or more old. Calibration stickers for its radiation monitors were not evident. The inspector was informed by licensee personnel that since the PASS was used only infrequently, regular calibration was not required and that a schedule would be established on the basis of experienced reliability.

Although the inspector was informed that some spare parts for the PASS were available, a list of them could not be provided by licensee personnel during the inspection.

B. Radiation Monitors

The value of and the basis for the alarm and warning set points of the radiation monitors (Eberline RIIA) could not be determined during the inspection. Also, initially after the radiation monitors are energized, a "normal" indication is illuminated. However, in low background fields, it disappears shortly thereafter (due to the infrequency of the pulses which trigger it).

C. PASS Panel Indications

The licensee's procedures for the operation of the PASS (831.10) instruct the operator to verify that selected illuminated valve and pump status indicators on the PASS panel logic diagram have energized or de-energized. Other steps which also cause a change in one or more indicators that could be useful for diagnostic purposes are not called out in the procedures.

4.2.5.1 Recommendations for Improvement

A. Revise Procedure 824.9 to address lower limit of detection capability.

B. Ensure PAS-cask has been maintained in accordance with Quality Assurance Program for transport packages prior to use.

C. Complete site specific Core Damage Estimate Procedures.

D. A defined calibration and maintenance schedule should be devised.

E. A spare parts inventory should be documented.

F. The basis for the alarm and warning set-points of the PASS radiation monitors should be documented. Also, the procedures for the PASS should specify that the operator observe that they are operational when the PASS is initially energized.

G. To aid operators, the proper indications of the lights on the PASS control panel logic diagram should be called to the operator's attention at appropriate procedural steps where they should be energized or de-energized.

This item will be reviewed during a subsequent inspection (219/86-01-02).

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

5.1 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 3.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 3.

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

5.2 Findings

Within the scope of this review, the following was identified:

5.2.1 System Description

The licensee purchased and installed two Radioactive Gaseous Effluent Monitoring Systems (RAGEMS) supplied by Science Applications Inc. (SAI), one was to monitor the effluent released from the plant stack and the other was to monitor the effluents from the turbine building. They were originally designed and intended to perform monitoring and sampling of plant effluents in routine concentrations. Due to technical problems, they never became fully operational. Following the promulgation of NUREG-0737 functions the system was modified by the licensee to perform the high range functions.

The original system had been designed to perform continuous on-line analysis of integrated samples of radioparticulates and radioiodines and to continuously monitor and analyze for radio gases, so as to determine the precise amount of each isotope released. It includes three stages; a particulate filter, a halogen filter and a noble gas channel. They were arranged in series with three high-purity germanium detectors (HPGE) to perform the analyses. Both systems are controlled by one PDP-11/34 computer with a central terminal for readout of data.

The licensee modified the system by deactivating the HPGE capability and switched over to assessment of the noble gas activity using an ion chamber viewing the 6000 cc sample volume.

5.2.2 Findings

- Credit for dilution has been assumed in the licensee's contention that an upper range of 10^3 µCi/cm^3 (plant vent) is sufficient to meet the requirements of II.F.1-1. However, the licensee has not demonstrated this concentration could not be exceeded.

- It was not demonstrated that the installed high and low range monitors can provide range overlap.

- The only calibration of the ion chamber that has been performed to date has been for one point, using Xe-133 gas. An upper range of 10^3 µCi/cc was extrapolated from that point. Currently the data obtained from the ion chamber, independent of the time after shutdown, is reported as Xe-133. The energy response function of the detector for higher photon energies has not been determined. Those responsible for using the data for dose assessment are not cognizant of the calibration method.

- High concentrations of post-accident Noble gases may "burn-out" the photomultiplier tubes of the low-range detectors, so that the system would not be able to follow subsequent decreases.

- It has not been demonstrated that the stack low range monitor is adequately shielded from cross-talk, due to other possible local radiation sources (see Section 6.0 for detailed description of other sources) such as by-pass filters, unshielded piping in the monitoring shack, shine from the adjacent main stack, etc.

- The turbine building RAGEMS does not include a low-range monitor.

- Only limited training on the operation and readout of the RAGEMS noble monitor has been provided. Currently only four persons are trained to query the computer terminal for data. During off hours a delay up to an hour is possible before trained personnel would be available to obtain data from the system.

- Routine calibration and maintenance of the RAGEMS have not been implemented, nor have procedures been developed.

5.3 Acceptability

Based on the documentation discussed during the inspection, the installed system does not meet the requirements for high range noble gas monitoring as contained in NUREG-0737, Attachment II.F.1-1. Further documentation and/or improvements are required as follows:

A. The licensee should demonstrate that the current upper range capability of the installed gas monitors would not be exceeded in a worst case accident.

B. Calibration over multiple decades using transfer sources of varying energy should be performed. The results should be incorporated into the dose assessment function.

C. A low range capability should be installed on the turbine building monitor or it should be demonstrated that it is not required.

D. The overlap of the high and low range monitors should be demonstrated.

E. A method to deactivate the low range monitor near the upper bound of its dynamic range and to reactivate it when the high range monitor returns to the low end of its range should be devised.

F. A study on the effects of other nearby radiation sources on the response of the low range monitor should be made.

G. Additional personnel should be trained so as to provide 'round-the-clock" readout of the effluent monitors or a simple readout should be provided to the control room operators.

H. Routine calibration and maintenance procedures should be provided and training to the operational personnel be accomplished, such that normal surveillance of the RAGEMS will be performed.

This item is considered unresolved and will be reviewed during a subsequent inspection (219/86-01-03).

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

6.1 Position

NURGE-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. It must be performed 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 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 3.0 and in regard to licensee letters, memoranda, drawings and station procedures as listed in Attachment 3.

The licensee's performance relative to these criteria was determined by interviewing the principal persons associated with the design, testing, operation 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 systems.

6.2 Findings

Within the scope of this review the following was identified:

6.2.1 System Description

Sampling of particulates and iodines is performed sequentially in the first two stages of RAGEMS. Both stages can be manipulated remotely from the computer terminal when filter changes are required. However, entry into the sampling shack is

required to initiate sampling thru RAGEMS and later to retrieve the filters after they have been automatically ejected from the sampling position. According to the licensee procedures (406.6) RAGEMS would be placed in service by a change in inlet valve lineup only in the event of a high indication (> 10^5 cps) of the normal gas monitor. RAGEMS is not presently used for continuous sampling. RAGEMS is not used for routine sampling and a flow of 1.5 CFM is routed through an unshielded by-pass particulate and iodine filters, so as to prevent excessive amounts of activity to accumulate on the filters and thus making retrieval and isotopic analysis suspect with regard to exposure constraints. It is the licensee's plan, in the event of an accident, that readily retrievable filters will be placed on-line for two seconds, and then retrieved for analysis. Given the sample flow rate and the maximum concentration assumed by the licensee of approximately 5 μCi/cc, isotopic analysis of filter cartridges installed on a shielded and collimated holder, can be performed.

The 1.5 CFM sample flow is isokinetic and the system has an active means of adjusting the flow rate to account for changes in stack flow over a limited range.

Findings

- The licensee's capability to shield, transport and analyze radioiodine particulate and gaseous samples within the design basis range specified in Table II.F.1-2 is dependent on the assumption that the plant effluents will be significantly less than 100 μCi/cc.

- Methods, training or procedures to perform representative sampling of iodines and particulates, in accordance with Table II.F.1-2 were not demonstrated. The proposed two second sample time appears inadequate to provide a representative profile of the stack concentrations at the time of sampling since it is doubtful that the flow through the filter and sampling pipe would reach equilibrium in this short interval. Possible sources of error include insufficient purge of air in the sample time due to the lack of correction for valve opening and closing. It does not appear adequate to collect a sufficient sequence of two second samples to meet the requirements of II.F.1-2 for continuous sampling.

- Variations in plant parameters that could cause fluctuations in stack flow are beyond the dynamic range of the active flow control of the sampling system. Procedures for the resulting non-isokinetic flow condition, with appropriate corrections are not available.

- Entrained moisture could degrade the absorber under some postulated accident conditions, since the sampling lines are not heat traced within the sampling shacks, where the shack heaters serve as the heat tracing once the lines enter. In the event of loss of off-site power the building heaters are not connected to the vital power bus or reliable source of backup power, leaving the inside lines unheated.

- A comprehensive time and motion exposure study to demonstrate the sampling methods could be accomplished within the GDC-19 limits had not been made. A number of potential radiation sources were neglected in the study that had been performed but is in draft. The licensee had not considered the possible contribution of dose due to shine from the stack, unshield piping and water trap and the build up of high levels of iodines and in the bypass filter cartridges. For example, if a maximum value of 10 μCi/cc is assumed in the stack (a factor of 10 less flow is stated in NUREG-0737) and the sample flows through the bypass filter for 30 minutes prior to the sample retrieval, the dose at one foot from the bypass filter would be approximately 10 R in three minutes.

- Adequate procedures and training to retrieve and analyze iodine and particulate samples are not available.

- The licensee had not implemented an appropriate routine maintenance and calibration of the RAGEMS particulate and gaseous radioiodine sampling stages.

6.3 Acceptability

Based on the documentation discussed during the inspection, the licensee had not demonstrated that the installed system meets the requirements of NUREG-0737, Attachment II.F.1-2 and that samples can be obtained and transported within GDC-19 limits.

An evaluation of representative sampling capabilities whenever exhaust flow occurs must be documented and the required improvements completed as follows:

A. An appropriate site specific source term for release of radioiodines should be documented.

B. The sampling method should be redesigned to increase the sample time to provide a representative sample.

C. A procedure to apply appropriate correction factors during non-isokinetic conditions should be provided.

D. Heat tracing of the sample lines on vital power should be extended to the sample flow paths within the sampling shacks.

E. A comprehensive time/motion and exposure study to insure the GDC-19 criteria can be met for the retrieval and analysis of filters.

F. Appropriate procedures should be provided and the needed training of personnel conducted.

G. Routine maintenance and calibration of the particulate and gaseous radioiodine samples should be implemented.

This item is considered unresolved and will be reviewed during a subsequent inspection (219/86-01-04).

7.0 II.F.1-3 Containment High Range Area Monitor

This system has not yet been installed but the components have been purchased and some are onsite. The inspection consisted in a review of the design specifications and drawings, the manufacturer specifications, and discussions with project engineers. The system was found to conform to the requirements of NUREG-0737, II.F.1-3 in most respects. Some items could not be confirmed at the time of the inspection and are as follows:

-- Documents and tests to certify that the detector cables and junction connections in the drywell are environmentally qualified for drywell conditions during a postulated accident.

-- System drawings and layouts to verify that the proposed detector locations are not close to any equipment or piping that may contain radioactive fluids during an accident. Such components may cause interference in the detector's ability to respond to activity in the drywell atmosphere.

-- Verification of the nature of the signal sent out by the detector when the radiation field is below the lower limit of detection of the system. This signal is to be used to produce a failure indication upon detector failure(219/86-01-05).

8.0 II.D.3.3 Airborne Iodine Sampling During an Accident

The ability to sample for iodine and to count the samples during an accident were reviewed. The onsite assembly areas reviewed include the Operations Support Center and the Technical Support Center. Although the capability to collect samples during an accident appears to be adequate, some concerns were not resolved during the inspection and must be addressed in a later inspection. These items are as follows:

-- The ability to count the air samples collected. Questions in this area relate to the availability of sufficient counting systems to handle the expected large volume of samples, as well as the susceptibility of such systems to being disabled by high ambient radiation fields.

-- The exact lines of authority during an accident, including the mechanisms that would be used to initiate sample collection and the assignment of priorities in counting those samples during an accident (219/86-01-06).

9.0 Quality Assurance and Design Review

9.1 As part of the inspection effort a review was performed to verify and validate the adequacy of the licensee's design and quality assurance program for the installation of the Post-Accident Sampling System.

Documents Reviewed

The implementation, adequacy and status of the licensee's Post-Accident Sampling and Monitoring System were reviewed against the criteria identified in Section 3.0 and in regard to licensee correspondence, Specifications, Functional Tests, Vendor Drawings and station procedures as listed in Attachment 4A.

The licensee's performance relative to these criteria was determined by interviews with principal personnel associated with the installation and Testing of The Post-Accident Sampling System.

9.2 Findings

The Post-Accident Sampling System has been classified by the licensee as Nuclear Safety Related requiring installation in accordance with the GPU Nuclear QA plan. Sample piping up to and including the second isolation valve is designed and installed to seismic class 1 requirements. Sample piping beyond the second isolation valve is designed and installed in accordance with ANSI-B31.1 requirements. Electrical power to the Post-Accident Sampling System Control panel ER-19 comes from one of two thirty ampere circuits in distribution panel PNL-PD-8. Power to panel PNL-PD-8 is derived from the Safety Substation 1B2 through distribution panel "D". Panel PNL-PD-8 includes an undervoltage trip device to prevent re-activation of its electrical load on loss of off-site power, requiring manual activation to put the Post-Accident Sampling System back on line.

The Post-Accident Sampling System, a Generic BWR LOCA Sampling System, has incorporated all the changes/modifications identified by the manufacturer and users of similar equipment at other installations.

Within the scope of this inspection, no violations or unresolved items were identified.

10.0 Exit Interview

The Post Accident Sampling and Monitoring Team met with the licensee's representatives at the conclusion of the inspection on January 17, 1986. The Team Leader summarized the purpose, scope and findings of the inspection. Dr. W. Pasciak informed Mr. P. Fiedler during a subsequent telephone discussion on January 24, 1986, that the findings, as discussed during the exit meeting and as documented in Sections 5.3 and 5.6 of this Report are considered unresolved items.

At no time during the inspection was written material provided to the licensee.

Attachment 1

Oyster Creek Nuclear Generating Station Procedures

-- 823.1, "Chemical Analysis: pH"

-- 823.2, "Chemical Analysis: Conductivity"

-- 823.7, "Chemical Analysis: Boron"

-- 823.7.1, "Chemical Analysis: Boron"

-- 824.1, "Chemical Analysis: pH Meter"

-- 824.2, "Chemical Instrumentation Conductivity Bridge and Cell"

-- 824.6, "Chemical Instrumentation: Spectrophotometer, UV/VIS (Perkin Elmer Lambda 1)"

-- 824.8, "Chemical Instrumentation: Gas Chromatograph"

-- 824.9, "Chemical Instrumentation: Ion Chromatograph"

-- 826.1, "Radiochemical Instrumentation: Canberra Analysis System"

-- 831.3, "Post-Accident Sampling and Analysis Preparation and Analysis", Revision 4, dated November 25, 1985.

-- 831.9, "Post-Accident Sampling and Analysis PASS Analytical Program", Revision 1, dated December 12, 1985.

-- 831.10, "Operation of the GI Post-Accident Sampling", Revision 3, dated January 20, 1986

-- 831.11, "Post-Accident Sampling and Analysis Cask Transport Off-Site", Revision 6, dated November 26, 1985.

Oyster Creek Nuclear Generating Station Drawings

- P&ID 3431-M0012, "Flow Diagram Post-Accident Sampling", dated October 14, 1984.

Licensee Correspondence

- P. B. Fiedler, V.P. Nuc. GPU, to D. G. Eisenhut, Dir. DOL, dated April 20, 1982.

- P. B. Fiedler, V.P. Nuc. GPU, to D. G. Eisenhut, Dir. DOL, dated June 15, 1982.

- D. M. Crutchfield, Chief ORB #5, to P. B. Fielder, V.P. Nuc. GPU, dated June 30, 1982.
- D. G. Eisenhut, Dir. DOL, to P. R. Clark, Exec. V.P. GPU, dated July 30, 1982.
- D. M. Crutchfield, Chief ORB #5, to P. B. Fiedler, V.P. Nuc. GPU, dated October 10, 1982.
- P. R. Clark, Exec. V.P., GPU, to D. G. Eisenhut, Dir. DOL, dated December 24, 1982.
- D. M. Crutchfield, Chief, ORB #5, to P. R. Clark, Exec. V.P. GPU, dated January 17, 1983.
- P. B. Fiedler, Dir. DOL, to D. G. Eisenhut, Dir. DOL, dated April 15, 1983.
- P. B. Fiedler, Dir. DOL, to D. G. Eisenhut, Dir. DOL, dated May 20, 1983.
- D. G. Eisenhut, Dir. DOL, to P. B. Fiedler, V.P. Nuc. GPU, dated February 10, 1984.
- P. B. Fiedler, V.P. Nuc. GPU, to D. M. Crutchfield, Chief ORB #5, dated March 6, 1984.
- P. B. Fiedler, V.P. Nuc. GPU, to D. M. Crutchfield, Chief ORB #5, dated March 16, 1984.
- P. B. Fiedler, V.P. Nuc. GPU, to D. M. Crutchfield, Chief ORB #5, dated July 19, 1984.
- W. A. Paulson, Actg. Chief ORB #5, to P. B. Fiedler, V.P. Nuc. GPU, dated August 29, 1984.
- P. B. Fiedler, V.P. Nuc. GPU, to D. G. Eisenhut, Dir. DOL, dated October 3, 1984.
- P. B. Fiedler, V.P. Nuc. GPU, to D. G. Eisenhut, Dir. DOL, dated October 10, 1984.
- P. G. Loza, OC Fuel Projects Engineer to G. R. Bond, GPU Nuclear Analysis and Fuels Director, dated December 11, 1984.
- R. D. Gagliardo, Prog. Mgr., Burns & Roe, to D. N. Green, Mgr. OCNCS Eng. Proj., dated March 20, 1985.
- P. B. Fiedler, V.P. GPU, to J. A. Zwolinski, Chief ORB #5, dated April 22, 1985.

Other Correspondence

- T. A. Green, Mgr., Servicing and Aux. Equipment Retrofits, GE Co., to T. H. Wyllic, Mgr. Brunswick Engineering, CP & L, dated April 6, 1984.

Vendor Manuals

- NEDO-24889, "Post-Accident Sampling System O&M Manual".

Attachment 2

Comparison of Chemical Analytical Test Results

Boron

Standard (ppm)	Dilution Factor	Meas. Conc. (ppm)	Analyses Results&Error(ppm)	Licensee Commitments(ppm)
997.7	100	10.2	1020 + 22.3	+/-50(0 to 1000)
2981	1	3065	3065 + 84	+/-300(> 1000)
4895	100	4.8	4800 - 95	+/-300(> 1000)

Chloride

Standard (ppm)	Dilution Factor	Meas. Conc. (ppb)	Analyses Results&Error(ppm)	Licensee Commitments(ppm)
10.04	1000	10.7	10.7 7%	+/-10%(0.5 - 20ppm)
30.11	10,000	3.7	37 <0.5 ppm	+/-0.05ppm(0-0.5ppm)
70.06	10,000	7.6	76 9%	+/-10%(0.5 - 20ppm)

pH

Standard	Analyses Result	Error	Licensee Commitment
6.198	6.3	+0.1	+/-0.3

Hydrogen

Standard	Analyses Result	Error	Licensee Commitment
3.2%	3.1%	.3%	+/-10%

Attachment 3

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

Oyster Creek Nuclear Generating Station Procedures

-- 831.7, "Post Accident Sampling and Analysis: Preparation and Analysis", Revision 4, dated November 15, 1985.

-- 831.4, "Post Accident Sampling and Operations: RAGEMS", Revision 2, dated November 22, 1985.

-- EPIP-9, "Off-site Dose Projections", Revision 5, dated January 17, 1985

-- TP-300/0.1 MTX 138.9.2.1, "Station 3 Ion Chamber Full Scale Range Determination", Revision 0, dated ?

Correspondence

- J. Knubel, Mgr. BWR Lic., to D. Crutchfield, Oper. Rec. Br. #5, DL, dated February 18, 1983.

- P. Fiedler, V.P. to D. Eisenhut, Dir. DOL, dated February 23, 1982.

- D. Muller, Asst. Dir. Rad. Prot., DSI to T. Novak, Asst. Dir. Lic. DL, dated October 9, 1984.

- H. Kister, Chief Proj. Br. No. 1, Div of Reactor Proj., to P. Fiedler, V.P. & Dir. OCNGS, dated March 27, 1985.

Licensee Internal Memoranda

- J. Stevens to File, dated July 23, 1985.
- G. Sadauskas to M. Laggart, Dated July 17, 1985.
- J. Stevens to B. Hohman, dated November 11, 1985.
- J. Stevens to Mgr. Lic., dated July 26, 1985.
- J. Cline to S. Gera, dated May 4, 1984.
- J. Stevens to S. Gera, dated July 9, 1984.

United Engineers Correspondence

- J. Ucciferro, Proj. Mgr., to D. Chandler, dated January 15, 1986.

Scientific Applications Inc. Documents

- Assessment of Radiation Dose Rate in the Oyster Creek Stack RAGEMS Building, James Cline, SAI Inc., Rockville, Maryland, GPU Nuclear Number 990-1214.

Attachment 4

Documentation for NUREG-0737, II.B.3

5.1 Vendor Manuals

1.1.1 Post-Accident Sample Station, General Electric Company No. NEDC-24889

5.2 Drawings

5.2.1 Burns and Roe Drawings

BR-M0012,	Revision 7, Flow Diagram Post Accident Sampling System
BR-E0172,	Revision 4, Miscellaneous Power Panels Post-Accident Sampling System - Power Distribution
BR-E0215	Revision 2, Miscellaneous Connection Diagram General Electric LOCA Sampler Panel ER-19
BR-E0366	Revision 1, Internal Wiring Diagram - General Electric LOCA Sampler Control Electric Panel ER-19. Field Change Request Nos. FCR-016469, 022503, 032919 & 032917.
BR-M0123,	Revision 2, Post-Accident Sampling ISO - Reactor Building Containment Atmosphere Sample Supply.
BR-M0124,	Revision 2, Post-Accident Sampling ISO Reactor Building Panel H 21T-A to TIP Room.
BR-M0126,	Revision 4, Post-Accident Sampling ISO Reactor Building Liquid Sample Return to Waste Treatment.
BR-M0127,	Revision 2, Post-Accident Sampling ISO Reactor Building Liquid Sample Return to Torus.
BR-M0129,	Revision 3, Post-Accident Sampling ISO Reactor Building Reactor Cooling Sample Supply.
BR-M219,	Revision 1, Post-Accident Sampling ISO Reactor Building Core Spray and Shutdown Cooling Sample Supply.
BR-M0246,	Revision 0, Type 48B Typical Tubing Supports.
BR-H0254,	Revision 2, Type 47A&B Typical Tubing Supports (120°F and up)

5.2.2 General Electric Drawings

GE-C5474-E-601, Revision 2, Generic BWR LOCA Sampler Electrical Schematic Diagram

GE-C5474-E-603, Revision 3, Generic BWR LOCA Sampler Electrical Connection Diagram

GE-C5464-E-607, Revision 0, Generic BWR LOCA Sampler Electrical Graphic Panel.

GE-C5474-E-101, Revision 2, Generic BWR LOCA Gas Sampler Mechanical Flow Diagram

GE-C5474-E-102, Revision 3, Generic BWR LOCA Liquid Sampler Mechanical Flow Diagram.

Field Disposition Instructions FDI No. 367-91700

5.2.3 Oyster Creek Generating Station Procedures

OCNGS Procedure No. 119, Revision 6, " Housekeeping".

OCNGS Procedure No. 120, Revision 10, "Fire Hazards".

OCNGS Procedure No. 112, Revision 15, "Oyster Creek Calibration of Maintenance Test and Inspection Tools, Gauges and Instruments".

5.2.4 GPU Nuclear Technical Functions Procedures

SP-001, Revision 0, "Startup and Test Program and Test Requirements".

SP-002, Revision 0, "Test Procedure Generation/Approval/ Change".

SP-003, Revision 0, "Turn Over From Maintenance and Construction and Test Performance".

5.2.5 Plant Modifications

1.2.5.1 BA-402048, Post-Accident Sampling Systems Phase II

5.2.6 Specifications

Installation Specification for Electrical Installation for Post-Accident Sampling System - Phase II, OCIS - 402048-005 No. 399.00-9, Revision 1.

Installation Specification for Post-Accident Sampling System Phase II (Mechanical) OCIS-399-11, Revision 2.

Division II, System Design Description for Post-Accident Sampling System, SDD-OC-555, Revision 0.

5.2.7 Technical Specification

Oyster Creek Nuclear Generating Station Technical Specifications up to and including Amendment 74.

5.2.8 Test Procedures

TP 280/3, "Functional Testing for Post-Accident Liquid Sampling Valves using Clean Water".

TP 280/4, "Functional Testing for Post-Accident Gas Sampling Valves using Clean Gas".

TP 280/8 "Functional Testing of Post-Accident Sampling System Miscellaneous Solenoid Operated Valves".