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REGION I

Report Nos. 50-317/85-18, 50-318/85-16

Docket Nos. 50-317, 50-318

License Nos. DPR-53, DRP-69 Priority -- Category C

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

Facility Name: Calvert Cliffs Nuclear Power Plant Units 1 and 2

Inspection At: Lusby, Maryland

Inspection Conducted: July 15-19, 1985

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Summary: Inspection on July 15-19, 1985 (Report Nos. 50-317/85-18, 50-318/85-16)

Areas Inspected: Special inspection of: the operability status of the Post Accident Sampling System (PASS); the adequacy of alternate means of post accident sampling; PASS procurement, installation, and acceptance testing; installation and acceptance of other selected TMI Action Plan hardware modifications; and management and committee oversight of PASS.

Results: The PASS and alternate means of post accident sampling remain in a degraded condition. Uncertainties exist with the ability to obtain specific sample results. At the time of the inspection, the licensee was unable to perform a complete and accurate analysis of the reactor coolant in a post accident mode. Activities associated with PASS procurement, including installation testing and committee review, appear to have been adequate. Problems were identified with the management oversight associated with ensuring continued PASS operability. A significant misunderstanding was found to exist within the BG&E organization regarding the alternate method of post accident sampling to be used to meet Technical Specification Requirements.

The licensee submitted a corrective action plan at the Exit Interview, "PASS and Backup PASS Method Action Plan," which is included as Attachment 1 to this report.

DETAILS

1.0 Persons Contacted

Within this report period, interviews and discussions were conducted with various licensee personnel, including chemistry department technicians, quality assurance representatives, training instructors, and the licensee's management staff.

An entrance meeting was conducted with the licensee management on July 16, 1985 to outline the scope of the inspection.

2.0 Overview of Inspection

2.1 Purpose of Inspection

This inspection is a follow-up to that conducted by Mr. John White during the period June 24 to June 28, 1985. (50-317/85-16, 50-318/85-14). That inspection revealed significant problems associated with the implementation of NUREG 0737 items dealing with Post Accident Sampling. As a result of that inspection, a Management Meeting was held with BG&E personnel in the Region I Office on July 11, 1985. At that meeting, the licensee was informed that a followup inspection would be conducted as a result of significant NRC concerns.

This followup inspection had a two fold purpose: first, to determine the operational status of the PASS as well as what factors contributed to the continuing technical problems of the system; second, to determine if the problems found to exist with PASS installation and operation extend to other systems, particularly those associated with the implementation of NUREG 0737 items.

2.2 Conclusions

The PASS and alternate means of post accident sampling remain in a degraded condition. The deficiencies associated with PASS appear to be an isolated case and no significant deficiencies exist with similar NUREG-0737 items.

A long history of PASS technical problems and inoperability is a result of a number of factors. The only dominant factor here appears to be the lack of strong managerial oversight to resolve these problems in an effective manner. A review of the implementation of other NUREG-0737 items failed to reveal similar problems; therefore, it is concluded that the problems with PASS are an isolated occurrence. A list of the factors which contributed to continuing operational problems follows.

- Combustion Engineering (CE) was considered by the licensee to be the sole source at the time of the systems procurement. In hindsight, BG&E management considered the system to be more complicated

than needed since NUREG-0737 does not require an in-line analysis capability. Operability of in-line measurement devices (H_2, O_2, Ph, Boron) have had a high incidence of failure.

- From June 1982 to January 1984 the System was never used to sample directly from the RCS. No training or surveillance was conducted during this time because the system concept was that it was to be used only during a one time accident scenario. There was a desire to avoid internal contamination of the system to make maintenance easier.

- After January 1984, when a sample was drawn from RCS due to urging of Resident Inspectors, the number of Maintenance Requests associated with system operability increased substantially, 2 MR's from 6/82 to 1/84 to 9 MR's from 1/84 to 7/85.

- Training & Procedure development was hindered by system unavailability due to hardware problems. The system was not designed for frequent operation. Leaky Valcor valves contributed to system availability problems.

- The system was declared operable (June 1982) without adequate testing and procedures in place since:

  a. Reactor Coolant sample not run until early 1984.
  b. Dilution accuracy for PASS GRAB samples was not verified.
  c. Accuracy of the Isotopic analysis was not evaluated until 2/85.

- Maintenance on PASS and other NUREG-0737 related systems, which do not operate on a daily basis, are given lower priority. Many of these systems "belonged" to the Chemistry Department not Operations, hence the lower priority.

- Licensee opted to put T.S. in more restrictive Section 3 as opposed to Section 6 as recommended in Generic Letter 83-37. The licensee proposed that an "alternate" method of drawing a PASS sample be put in the T.S. NRR concurred since this was a more conservative specification regarding system operability. There has been confusion concerning this "alternate" method. NRR & BG&E Engineering considered the alternate method to be PASS GRAB samples, Plant Superintendent and Operations considered it to be grab samples from the NSSS Sink. The NRC SER does not address the use of the NSSS sample sink.

- POSRC review of T.S. did not uncover the discrepancy of the "alternate" sampling method. Discussion with committee members indicates that the committee considered the "alternate" method to be the NSSS Sink.

- Weak Management Handling of PASS Operability.

  -- Little dialogue with other utilities with similar systems (St. Lucie/San Onofre). BG&E declined to participate in MATRIX test by CE at St. Lucie.

  -- Chemistry Department was not aggressive in resolving system deficiencies and emphasizing system operability. When the system came under T.S. in 2/85, only piecemeal surveillance was conducted. Problems with Isotopic analysis, although identified, were not pursued to resolution.

  -- Full, integrated test of the system was not attempted until June 1985, shortly before the initial NRC inspection.

  -- Management intervention, above the level of Chemistry Supervisor, was needed to set in motion a plan to resolve PASS problems in a timely manner (week of 7/15).

3.0 Review of Technical Problems Associated with Operability of the Post Accident Sampling System (PASS)

There are three methods to draw a post-accident Reactor Coolant System (RCS) sample:

PASS in line analysis
PASS diluted grab sample
NSSS primary sample sink undiluted grab sample

At the completion of this inspection, the accuracy or operability of all these methods were in question for various reasons. No accurate, reliable method was available to determine, with confidence, post-accident reactor coolant radio-chemical analysis parameters.

3.1 The following performance tests were made:

Chemical Analyses

Blind tests of licensee's laboratory analytical capability were performed for boron and chloride prepared test solutions provided by the inspector.

Results

Boron Test Solution	Chloride Test Solution	Licensee Analysis	Acceptable Tolerance
2981 ppm		2966 ppm	±5%
	30.11 ppb	31.2 ppb	±10%

The licensee was able to demonstrate acceptable laboratory analytical ability for these two post accident sampling parameters.

3.2 Combustion Engineering Post-Accident Sampling System (CE-PASS) Grab Sampling

On July 18, 1985, the licensee demonstrated the grab sample capability of the CE-PASS skid. Samples were drawn from both the Unit-1 Low Pressure Safety Injection System and the Unit-2 Steam Generator Hot Leg. These samples demonstrated that fluid was able to pass through the skid, that the components affecting dilution were operable, and that a reactor coolant sample could be acquired.

Although the system demonstrated that a sample could be drawn, its dilution capability was found to be inaccurate. In this condition, the system is unable to provide a known dilution factor which is necessary to adequately quantify isotopic and chemical constituents. The licensee has initiated action to resolve problems associated with determining a known dilution constant as part of the action plan for improving post accident sampling capability.

The following should be considered for grab sampling via this mode:

-- the installation of a drip pan under the sampling septum to prevent inadvertent contamination of the area;

-- increased shielding between the in-line radioisotope analyzer volume chamber and the grab sampling septum and operating panel;

-- the installation of an area radiation monitoring device in the CE-PASS operating area; and,

-- the installation of a shielded assembly, including high integrity shielded syringes to effect acquisition of undiluted reactor coolant sample for back-up pH and total/hydrogen gas analyses.

3.3 Nuclear Steam Supply System (NSSS) Sink and Post-Accident Sampling Apparatus (PASA)

The following applicable procedures were reviewed and field tested in an NRC-observed exercise of the licensee's emergency response plan.

ERPIP Rev. 10	4.4.7.3	Post Accident Reactor Coolant Sampling
ERPIP Rev. 10	4.4.7.4	Post Accident Reactor Coolant analysis

The actual use of these procedures in an accident situation is unlikely since personnel access to the NSSS Sinks would probably be prohibited due to high radiation. Additionally, the use of the NSSS Sinks as a post accident sampling alternative has not been reviewed by NRR relative to NUREG-0737 requirements, nor does the current safety evaluation address NSSS Sink use for any type of post-accident sampling.

However, during this inspection effort, the licensee was in the process of performing a time and motion study to determine if the method could be performed within GDC-19 criteria.

The licensee attempted several times to demonstrate operability of the system including the Post Accident Sampling Apparatus (PASA).

The following was noted:

-- In a performance demonstration on July 17, 1985, the PASA leaked all of the collected reactor coolant through a hose clamp connection into the chemistry hood. This hose clamp connection was a repair to the system made subsequent to a previous NRC identified deficiency.

-- In a follow-up demonstration on July 18, 1985, upon improving the connection and using a new burette, the PASA again leaked the majority of the reactor coolant sample into the chemistry hood. It was later discovered that the new burette stop-cock assembly was of such design to prevent controlled delivery and isolation of the reactor coolant sample in the burette. The chemistry technician had to hold his finger on the spigot to prevent complete sample loss.

-- Portions of the PASA assembly are constructed of glass which might be subject to breakage in a post accident condition. During a maintenance operation on July 18, 1985, the burette was broken in place, rendering the system inoperable.

-- A valve manipulation required to vent the sample collection bomb for sample delivery to the burette, was not addressed in procedure ERPIP 4.4.7.4. As a result, the sample could not be removed from the bomb until the valving error was rectified.

Throughout the course of this inspection the licensee was never able to satisfactorily demonstrate that the NSSS Sink and associate PASA would provide an adequate, reliable and viable option for post-accident sampling. In an actual post-accident condition, the findings noted could have a substantial effect on personnel exposure and result in significant contamination of facilities and equipment.

In the course of this review, the licensee's preliminary data to support personnel exposure consideration was examined. The data suggests a source term configuration that appears unrealistic and appears to underestimate actual personnel contact with highly radioactive components and processes. The licensee's data did not indicate that the operation could be done within GDC-19 limits relative to personnel extremity exposure. Further, the study conducted to analyze sampling from the NSSS sink only considered the first sample, a condition in which the NSSS Sink is initially unaffected by a high source term. Subsequent sampling from the sink, in which a constant source term is involved, was not considered.

3.4 Personnel Training

The licensee has initiated a technical task force comprised of chemistry, radiation protection, training, emergency planning, and operations personnel to develop and field test post accident sampling procedures, incorporate such procedures into emergency planning implementing procedures, and to develop training lesson plans. At the time of this inspection, the task force was pursuing efforts to use the NSSS Sinks and associated PASA systems. A training program relative to familiarization with the equipment and apparatus had previously been completed for all chemistry technicians that might be assigned to post accident sampling tasks.

No other training efforts were in progress at the time of this inspection.

4.0 Licenses Management Oversight

4.1 System Operability

PASS was declared operable on June 1, 1982 without a final operating procedure in place, with only a limited amount of system training being given to the users (plant chemistry personnel), and without taking an actual RCS sample. Actual RCS sampling was not done due to an overriding concern by maintenance groups that such sampling would contaminate the PASS system and thereby cause increased radiation exposure to maintenance personnel. Analyzers in the system were tested by substitute means (test solutions, electronic calibrations, etc.). As a result of the decision not to run RCS fluid through the system, comparison of results between PASS and the normal sample sink samples was not performed. Failure to perform this test prevented an early identification of equipment problems which only became evident when actual samples were later drawn. The initial acceptance testing also failed to verify the accuracy of dilutions done within the PASS system.

Improvements have been made by the licensee in providing management oversight of the "systems turnover" for operations process. Where appropriate, Calvert Cliffs Instruction CCI 126E, dated May 24, 1984, requires that the General Supervisor, Operations conduct a review and determine if a new or modified system is acceptable and operational. This judgment is based upon completion of a system walkdown, the updating of critical drawings, implementation of associated Technical Specifications, revision of operating procedures, and completion of training. This was not required in 1982 at the time of PASS implementation. CCI 126 currently requires the POSRC to do a final review of safety-related Field Change Requests (FCR's) and test results. The Plant Superintendent does the final review of non-safety related FCR's and their test results. PASS, although non-safety related, was reviewed by POSRC in 1982 because it was a TMI action plan item.

Once declared operable, no overall responsibility for ensuring system operability was assigned. This is not unusual. Typically no such responsibility is assigned for operable plant systems. In the case of PASS,

however, where many latent problems existed after system acceptance, this type of assignment may have been useful. Early in 1985, the Plant Superintendent did charge the General Supervisor, Chemistry with responsibility for overall PASS system operation.

4.2. Operational Experience

In early 1984, at the urging of the NRC and INPO, the licensee decided to run actual RCS samples with PASS. Comparisons of PASS and normal sink samples were still not conducted. Since the time the system was declared operable, personnel training and procedure development have been hampered by system unavailabilities.

As equipment problems were noted with PASS, maintenance requests (MR's) were initiated. The number of MRs significantly increased in 1984 (the year in which actual RCS samples were drawn). The governing procedure for generation of MRs, CCI 200I, dated April 1, 1985, establishes criteria for determining maintenance priorities. Before issuance of Technical Specifications (TS), PASS would, by that criteria, be placed in a lower priority category (Priority 3 of 4 categories with Priority 1 being the top priority). Technical Specifications for PASS were issued on February 22, 1985. Of significant note, other TMI Action Plan required equipment which has been installed but for which Technical Specifications have not yet been issued could potentially still receive maintenance on a lower priority basis.

4.3 Technical Specification Development and Compliance

The licensee chose to draft a PASS Technical Specification which was more restrictive than the sample provided by the NRC in Generic Letter 83-37. Unlike the NRC sample, their Technical Specification requires operability of a "pre-planned alternate" system in the event in-line PASS was not available. This inspection revealed a significant misunderstanding within the BG&E organization regarding what constituted the alternate reactor coolant sampling system.

In a letter to the NRC dated November 30, 1982, the licensee stated that in-line monitoring via PASS was the primary method of obtaining all but the chloride analysis for the reactor coolant system (RCS). The taking of diluted, depressurized reactor coolant samples and diluted reactor coolant off-gas samples via PASS, with analysis in an on site or off site lab, is described as the means for meeting the NRC criterion for a backup grab sample system. The fact that the PASS grab sample was the backup system described to and later accepted by the NRC (Safety Evaluation dated February 12, 1985) was not clearly understood by the on site chemistry group, the Plant Superintendent and the Plant Operations and Safety Review Committee (POSRC).

Because of the misunderstanding noted above regarding what constituted the backup reactor coolant sampling system, the POSRC approved the draft Technical Specification for PASS believing that the alternate means for sampling referred to in the Technical Specification was a primary sample sink grab sample. This had been the system used by the licensee to fulfill interim NUREG-0578 RCS sampling requirements. After Technical Specification implementation the chemistry group realized that the interim 0578 system could not fulfill all sampling requirements of the PASS in-line system. Efforts were then initiated to upgrade the 0578 system. At the time of this Technical Specification implementation, and at the time of this inspection, the PASS system (in-line) was inoperable and the sample sink backup method appeared inadequate. Additionally, the true backup system, PASS grab, was inoperable in that the dilution accuracy had not been verified by the licensee. This inspection clarified which system had been reviewed and accepted as the primary (PASS in-line) and alternate (PASS grab systems).

The licensee currently intends to correct PASS in-line and PASS grab sample system discrepancies (dilution factor confirmation and Germanium detector efficiency factor determination) by July 31, 1985. Procedure and training enhancements will be completed by August 31, 1985. In the long term they plan to upgrade the PASS system (6 to 12 months) to accomplish and determine if PASS in-line and PASS grab are the optimum choices for primary and alternate sampling methods.

5.0 Committee Reviews of Proposal Plant Modifications

The inspector examined the records of the licensee's review committees with respect to the PASS and Containment Water Level Monitor modifications. The three committees involved are the Plant Operating Experience Assessment Committee (POEAC), the Off-Site Safety Review Committee (OSSRC), and the Plant Operations and Safety Review Committee (POSRC). The inspector verified that the committee reviews required by the Calvert Cliffs Technical Specifications (sections 6.5.1, 6.5.2 and 6.8.2) and the Calvert Cliffs Instructions (CCI-103H, CCI-126E and CCI-139D) were documented complete.

5.1 Overview of Committee Activities

The POEAC records indicated that experience from other facilities was being incorporated into plant operations. One identified example was found in the POEAC minutes for meeting number 85-12 held on June 20, 1985, which called for a review of PASS procedures for adequate flushing/draining requirements. The initiating item was a NETWORK entry and the committee designated a reviewer to be responsible for accomplishing the review. No deficiencies were identified.

The inspector reviewed the Facility Change Requests (FCRs) associated with the PASS and the Containment Water Level Monitor. The OSSRC reviews were contained in the FCRs as required. Each FCR package contains a list of POSRC meeting numbers at which the FCR was reviewed. The inspector

reviewed a sampling of the meeting minutes for both the OSSRC and the POSRC and verified that the committees had documented their review of the FCRs at the selected meetings. No deficiencies were identified.

5.2 Committee Review of PASS Technical Specification

The PASS technical specification, FCR 82-177, was reviewed by the POSRC and recommended for approval at POSRC meeting 82-136 held on October 27, 1982. During this meeting the committee reviewed 23 FCRs, 42 other documents, made changes to the committee's outstanding items list, discussed a recent plant event, and discussed a technical memo received by the POSRC chairman. All other POSRC minutes among the sample selected by the inspector exemplified a similar high volume of review work by the committee in that large numbers of documents were recommended by the POSRC for approval at each meeting.

The high volume of documents appears to indicate that each item receives minimal scrutiny by the POSRC as a committee. This places a high reliance on the working-level responsible engineer and the engineer's supporting line organization. This potential concern was presented to the licensee at the exit meeting. A similar concern was the basis for one of the recommendations contained in the most recent SALP Report which was transmitted to the licensee on January 18, 1985. The SALP Report recommended that the licensee "should assess whether the POSRC is organized in an effective manner such that adequate time is allocated to consider safety issues." The resident inspectors will continue to monitor the conduct and performance of POSRC reviews on an ongoing basis.

6.0 Review of the Field Change Request Process

The FCR process is presented in CCI-126E. The inspector reviewed the procedure with emphasis on the training of personnel and the revision of procedures to reflect system changes accomplished by the FCR. Each FCR package is required to include a completed checklist from each station organization potentially affected by the FCR. Each checklist is in the form of a memo to the head of one group (e.g., General Supervisor - Electrical and Controls) from the responsible engineer. By completing, signing and returning the checklist, each group head attests that the FCR has been reviewed and that any changes to lesson plans, procedures, manuals, etc. in that group's cognizance have been completed. The FCR package is not complete until all checklists have been completed and returned to the responsible engineer. The inspector reviewed the completed checklists in the FCR packages for PASS and the Containment Water Level Monitor; no deficiencies were identified.

7.0 Procurement, Installation and Acceptance Activities Associated With Selected 0737 Items.

The licensee's procurement, installation and system acceptance activities associated with the design changes for selected NUREG 0737 items were reviewed using the following Facility Change Requests (FCRs):

FCR 79 - 1057 Containment Wide Range Radiation Monitor (Item II.F.1 Attachment 3)

FCR 79 - 1058 Wide Range Noble Gas Monitor (Item II.F.1 Attachment 2)

FCR 80-1001 Wide Range Containment Pressure Monitor (Item II.F.1 Attachment 4)

FCR 80-1005 Hydrogen Sampling System (Item II.F.1 Attachment 6)

FCR 80-1008 Post Accident Sampling System (PASS) (Item II.B.3)

7.1 Engineering Review and Procurement

FCRs 80-1001 and 80-1005 above were to be designed as safety related in their entirety. The other two systems and PASS were not considered safety related downstream of the containment isolation valves per the guidelines of NUREG 0737. The above listed monitors were to be operational by January 1, 1985 according to the original NUREG 0737 guidelines. At the time of the issuance of NUREG 0737, the industry did not have commercial systems meeting all the guidelines of NUREG 0737. As a result, specifications and technical requirements for the above systems were to be developed. The industry was also faced with the difficulty of developing instrument ranges that could not be proof tested at the factory. For example, NUREG 0737 requires the containment high range radiation monitor to monitor up to 10^8 R/Hr. However, the industry did not have a test source to proof test the monitor to the specified upper limit. At a later date, NRC clarifications indicated that electronic calibrations would be adequate at these high levels. In order to meet the strict implementation deadlines and special technical requirements, the licensees worked with their respective nuclear steam supply system vendors and selected equipment vendors. Baltimore Gas and Electric Company worked with Combustion Engineering to develop the Post Accident Sampling System and three separate vendors for the remaining items. Because of the above, the licensee did not use the routine procurement procedure which requires detailed bid specification requests, bid evaluation and vendor selection. The licensee's engineers worked closely with the vendors to develop the technical requirements and design features. Integrated system testing was not possible at the factory, as the monitors and other components were to be placed in existing plant systems and controlled from the control room or other panels. A review of the licensee's activities during the developmental stage of the above systems indicated that the licensee worked closely with the vendors and provided direction and guidance on a continual basis. The responsible engineer and the QA representatives periodically visited the factory to assure that the equipment was manufactured in accordance with the specifications.

7.2. Design Activities

The system design features were reviewed by the inspectors with the responsible engineers against the guidelines of NUREG 0737 and the statements of the licensee's letter dated November 30, 1982. The responsible engineers were able to demonstrate how the guidelines of NUREG 0737 were factored into the system design. However, the backup documents to support the statements of the commitment letter were not readily available. The licensee's representatives stated to the inspectors that the NUREG 0737 modifications predated the licensee's current design change program which requires detailed documentation to support design activities. As a result, the supporting documentation for these modifications are not complete or readily available. The inspector informed the licensee's representatives that the NRC post implementation reviews of these modifications are still pending and these reviews may not be possible if the supporting engineering documents are not readily available. The inspector noted that in 1982, the NRC Performance Appraisal Team identified concerns in the licensee's safety evaluations for FCRs. In response to these concerns, the licensee revised the corporate and site procedures for FCRs. The inspector reviewed Revision 38 to QAP 15, the corporate procedure for Design changes and station procedure CCI-126 E for FCRs. These procedures provide adequate controls for the FCRs. The inspector reviewed an FCR in progress (FCR 81-1052) and noted an improved documentation level for this FCR. The quality and control of design inputs were better than those for earlier NUREG 0737 related modifications.

7.3 Quality Assurance

The inspector reviewed the following Quality Assurance audits in the Design Change area:

QAG 61-82-EED-10

QAG 61-84-18

QAG 61-83-15

QAG 61-84-23

For the most part, these audits were conducted to verify compliance with licensee procedures. Hardware and safety perspectives were not apparent in these audits. The licensee's QA representatives indicated that they were aware of the programmatic nature of the audits and were implementing steps to include performance orientation as a part of the audits. These steps included hiring experienced engineers to conduct QA audits and implementing the Paper for Technical Audits issued by the American Society for Quality Control.

The effectiveness of the licensee's audits for contractors was assessed by reviewing the following audits:

QAG-Bechtel 85- Program 10

QAG-Bechtel 84- Program 14

QAG-Bechtel 83- Program 08

QAG-CED 85 - Design 01

QAG-CED 84 - Design 01

QAG-CED 83 - Design 01

QAG-59-VEW- Surv 01

These audits contained detailed discussions of the review of technical items such as calculations, technical requirements and construction/assembly of equipment.

The inspector discussed the QA involvement in determining the effectiveness of the revised site and corporate FCR programs with QA representatives. Review and approval of the QAP for FCRs and periodic programmatic audits were the main QA involvement in the FCR program. The inspector discussed with the site and corporate engineering personnel and QA representatives the need to: 1) establish the level of compliance of the present FCR program with ANSI N45.2.11 and 10 CFR 50 Appendix B, Criterion III, and 2) establish the level of compliance of NUREG 0737 related modifications with the guidelines of NUREG 0737. These licensee representatives recognized the importance of the above need and agreed to take the required actions. Licensee actions will be reviewed during future inspections.

7.4 Conclusions

Except for the apparent lack of documentation to establish the adequacy of the design details, the inspector found that the licensee's design inputs were generally adequate to meet the design requirements specified in NUREG 0737 guidelines. The licensee's bid specifications and technical requirements were responsive to the guidelines of NUREG 0737. The licensee worked closely with the vendors to assure that the purchased equipment was designed per specification and was capable of meeting normal operation and accident environmental conditions. The installations were monitored and controlled by the licensee's engineers and QA/QC personnel. Pre-operational tests were designed to verify the design input requirements. These tests were performed adequately under the surveillance of engineering and QC personnel. Preventive maintenance and surveillance test requirements were adequately specified in design documents and vendor manuals.

8.0 Documentation Reviews of Environmental Qualification Issues Associated With Selected NUREG 0737 Items

8.1 Item II.F.1, Attachment 6 Containment Hydrogen Monitoring System

Calvert Cliffs uses Comsip Delphi Model K III hydrogen analyzers for post accident containment hydrogen monitor. These analyzers use the usual conductivity approach to measure the hydrogen concentration in percent volume. One analyzer is provided for each unit. Six sample points are provided at various locations in the containment. Samples are drawn sequentially from each of these sample points through an automatic sequencer and various sample line solenoid valves. These solenoid valves also serve as containment isolation valves. Room air where the analyzer is located is used for purging.

The environmental qualification of the Comsip Delphi hydrogen analyzer was described in EQ file HSAS01. The analyzer used for the qualification test was a Model KIV analyzer. BG&E used "similarity" method to qualify the Model K III analyzer. The differences between these two models were delineated and justified (where applicable) in attachment 1 to the "Qualifications Report Summary". Model K III used ASCO THT8262C7N and THT8262A138N solenoid valves while the tested model used THT8262C7E and THT8262A138E solenoid valves. The series "N" solenoid valves used BUNA N rubber for their seals and discs while the series "E" valves used ethylene propylene rubber. BUNA N is more susceptible to radiation and thermal aging. The licensee replaced the original solenoid valves (series "N") with series "E" ones. The valve replacements and post-installation test and verification were all documented in FCR No. 84-1020.

The inspector reviewed the EQ file and the valve replacement documents and found that the EQ for the installed hydrogen analyzer was acceptable.

8.2 Item II.F.1, Attachment 4 Containment Pressure Monitoring System

The containment pressure monitoring system uses three pressure transmitters (PT-5307, 5308, 5310) to measure the containment pressure. PT-5308 is for narrow range (-5 to +5 psig) while PT-5307 and PT-5310 are for wide range measurements (0 to 150 psig and -5 to 150 psig respectively). All of these transmitters are located outside the reactor containment and therefore not subject to a post LOCA, containment environment.

Two existing pressure transmitters (PT-5307 and PT-5308) are powered by the channel B bus. A third transmitter (PT-5310) was added to the channel B bus to meet NUREG-0737 requirements. The sensing taps (containment penetrations) for channels A & B are approximately 180 degrees apart.

A Sigma Model 9262 pressure indicator is provided for each of the three pressure transmitters for continuous indication in the control room. An electrically isolated signal is provided from PT-5310 to the plant computer for a containment pressure printout, when required. This ap-

proach was used instead of a continuous recording as required by NUREG 0737. This exception was quoted by the licensee in response to a NRC letter for classification of system design.

The inspector reviewed the documents pertaining to the design and installation of the system. All of the documents were accessible and found to be properly prepared, signed and updated. No unacceptable conditions were identified in this area.

9.0 Environmental Qualification of Inaccessible Valves in the PASS

The licensee provided information demonstrating that the PASS valves, which are not accessible after an accident, are environmentally qualified to operate. Based on the classification of these valves (Category 3, Reg Guide 1.97, Revision 3, May 1983), the inspector concluded that the licensee had justified the operability of these valves within their designed operational environment.

10.0 Licensee Action on Previous Inspection Findings

(Open) Inspector Followup Item (85-16-12)

The inspector reviewed the environmental qualification of the containment high range radiation monitor coaxial cable assemblies. In applying the Raychem to seal thermofit sleeves, the connectors between the electrical penetrations and the coaxial cables, the licensee used electrical tape for shimming instead of the special shimming material used in the qualification model. This item remains open until the coaxial cable assemblies are environmentally qualified.

Similar problems may exist for the coaxial cable assemblies in Unit 2, which is currently at full power with limited access. Verification of Unit 2 cable assemblies will be performed during the next outage.

11.0 Exit Interview

An exit meeting was conducted on July 19, 1985 with licensee management. The inspection team summarized their findings and clarified the required corrective actions to be taken by the licensee.

The licensee responded to the team's findings and presented a schedule, enclosed as Attachment 1, (PASS and backup PASS Method Action Plan) indicating proposed completion dates for corrective action on the PASS deficiencies.

ATTACHMENT 1

PASS and Backup PASS Method Action Plan

1. Resolve with NRC:
   - The applicable exposure limits (GDC-19 or 25R)
   - The time limit for analyzing samples (3 or 24 hours)
2. Determine which backup PAS method to adopt based on outcome of item 1:
   - Modified primary sample sink backup PAS to reduce exposures for sampling and Plant Lab analysis (10/1/85), or
   - Existing primary sample sink backup PAS to draw sample and send out for analyses (9/1/85), or
   - Modified PASS grab sample and Plant Lab analyses (8/15/85).
3. Establish PASS dilution factor repeatability:
   - Correct valve leakage affecting dilution process (7/19/85)
   - Obtain ten diluted samples (7/21/85)
   - Evaluate accuracy of selected samples (7/22/85)
   - Verify calibration of sample vessel level indication versus CE curve (7/26/85)
   - Remove and check CV-5028 (4-way valve) for 4.7 ml delivery (8/1/85)
4. Establish Germanium detector efficiency factor:
   - Collect at least ten Unit 2 γ scans for all three shield positions (7/26/85)
   - Evaluate and determine detector efficiency (7/31/85)
   - Obtain vendor assistance for determining detector efficiency for high-high geometry on collimator (7/26/85)
5. Finalize proposed modifications to PASS (8/15/85) (considering new H2 & O2 analyzers, isolation valves, relief valves)
6. Design, purchase, and perform PASS modifications (6 to 12 months).
7. Continue PASS and backup PAS procedure and training enhancements (8/31/85).

8. Review validity of backup PAS sample and anaysis exposure calculations (8/1/85) (dependent upon item 2).

9. Determine PASS response to representative test matrix (8/20/85) and resolve any discrepancies.

Resources Dedicated To Resolution of Above

- Steering Committee for Monitoring Progress (Tiernan, Denton, Russell)
- General Supervision Chemistry
- Two Chemistry Technicians
- Chemistry engineer
- Design and modification engineers (as needed)
- Maintenance crafts (as needed on priority basis)
- CE technical respresentative (as needed)
  (on site - 7/22/85)
- Bland, Inc. (as needed)
- Training instructors (as needed)
- Emergency Response Technician (as needed)

Note: **Indicated dates are estimated completion times based on optimum schedules**