U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos.	92-23 92-23
Docket Nos.	50-352 50-353
License Nos.	NPF-39 NPF-85
Licensee:	Philadelphia Electric Company Correspondence Control Desk P.O. Box 195 Wayne, Pa 19087-0195
Facility Name:	Limerick Generating Station, Units 1 and 2
Inspection Period:	July 19, - August 29, 1992
Inspectors:	T. J. Kenny, Senior Resident Inspector L. L. Scholl, Resident Inspector

Approved by: Clifford J. Anderson 9/18/92

Clifford J. Anderson, Chief Date
Reactor Projects Section No. 2B

Inspection Summary: This inspection report documents routine and reactive inspections during day and backshift hours of station activities including: plant operations; radiation protection; surveillance and maintenance; and safety assessment/quality verification. Our findings and conclusions are summarized in the Executive Summary that follows. Details are provided in the full inspection report that has been sent to those listed in the cc list attached to the cover letter.

EXECUTIVE SUMMARY
Limerick Generating Station
Report Nos. 92-23 & 92-23

Plant Operations

The Units operated continuously at or near 100 percent power throughout this period. Unit 2 set a GE BWR record for continuous operation without a plant trip or a safety related reason to remove the unit from service for maintenance or repairs. There was only one reportable event for both units during this inspection period. (Section 1)

Surveillance and Maintenance

Maintenance testing of motor operated valves, using Valve Operation Test and Evaluation System (VOTES), continued. There was one problem noted while testing a valve in the Reactor Core Isolation Cooling (RCIC) System. A limiting condition for operation was not formally entered, however, the condition was corrected before the time limit expired. Although not a violation of operational procedures, an operability concern was raised and resolved. (Section 3)

Engineering and Technical Support

The concerns of Bulletin 92-01 and the Supplement 1 (Thermo-Lag concerns) were addressed expeditiously by PECo engineering. Followup to a previous concern relating to Residual Heat Removal System heat exchanger corrosion was implemented by the installation of corrosion monitoring systems. (Section 4)

Radiological Protection

A portal monitor alarm, witnessed by NRC personnel, was addressed promptly by PECo Health Physics personnel. Also, PECo brought to the attention of the NRC incidents, where contaminated equipment had been removed from the protected area. PECo health physics personnel took the necessary steps to return the equipment and prevent recurrence. (Section 5)

Safety Assessment and Quality Verification

Although not written in an individual section of this report, several good safety practices (indicated above in each of the areas) were performed by PECo this period. The operation of the Units was very good. The followup to NRC Bulletin 92-01 and the followup to previous corrosion concerns were prompt and thorough. Health Physics followup to NRC and other self identified concerns were prompt.

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DETAILS

1.0 PLANT OPERATIONS (71707)[1]

The inspectors conducted routine entries into the protected areas of the plant, including the control room, reactor enclosure, fuel floor, and drywell (when access was possible). During the inspections, discussions were held with operators, health physics (HP) and instrument and control (I&C) technicians, mechanics, security personnel, supervisors and plant management. The inspections were conducted in accordance with NRC Inspection Procedure 71707 and evaluated the licensee's compliance with 10 CFR, Technical Specifications, License Conditions and Administrative Procedures. During this period, the inspectors performed 4 hours of deep backshift inspections.

1.1 Operational Overview

At the start of this report period both Units 1 and 2 were operating at 100 percent power. Except for some minor power reductions for maintenance and surveillance activities both units operated at full power for the entire inspection period.

On August 2, 1992, Unit 2 set a record for the longest continuous operating run for any General Electric Boiling Water Reactor. The previous record of 423 days was held by Georgia Power Company's Hatch Unit 1. Unit 2 has operated continuously, with no reactor trips or any safety related conditions identified necessary to shut the unit down, since returning to operation following its first refueling outage. The unit is scheduled to be taken off line for a refueling outage in January, 1993.

1.2 Reportable Events

Unit 1

There were no reportable events on Unit 1 during this report period.

Unit 2

On July 28, 1992 at 9:27 a.m., during the performance of Surveillance Test (ST) procedure ST-2-076-601-2, "NSSSS-Outside Atmosphere to Reactor Enclosure Differential Pressure-Low Channel "B" Functional Test, an Instrumentation and Controls (I&C) technician inadvertently caused various primary containment and reactor vessel isolation control system actuations. The resulting closure signals to the primary containment purge supply and exhaust valves, the reactor enclosure equipment compartment exhaust and nitrogen block valves and the reactor enclosure heating ventilation and air conditioning system isolation was an Engineered Safety Feature ESF activation.

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[1] The NRC Inspection Procedures used as guidance are listed parenthetically throughout this report.

PECo reported the actuation via the Emergency Notification System (ENS) as required by 10 CFR 72(b)(2)(ii).

After the operators determined that the actuations were caused by the technician, the affected systems were returned to normal using general plant procedure GP-8, "Primary and Secondary Containment Isolation Verification and Reset." The affected systems were returned to service by 10:46 a.m.

The consequences of the event were minimal and there was no release of radioactive material to the environment as a result of the actuations. All systems functioned as designed.

The technician performing the ST failed to adequately self check his performance of step 6.1.2 of the procedure. He looked at what he thought was the "proper switch" and saw it was in the RESET position, however, he was looking at the "Refuel Floor Ventilation System" switch rather than the "Refuel Area Secondary Containment Integrity" (Zone III) switch. The switches are located one above the other and are virtually identical. Because the proper switch for the ST was not in reset, when another technician at the logic panel continued the performance of the ST, the ESF actuation resulted.

A contributing factor to the event is that an alarm window on the main control panel in the control room is in alarm when the refuel floor Heating Ventilation and Air Conditioning (HVAC) switch is in reset (which is normally the case). Had this annunciator not been alarmed it may have served as an additional barrier to alert the technician of the proper switch in the reset condition.

The technician was counseled on the importance of attention to detail and self checking. A temporary change to the alarm annunciation capability, enabling the operator to clear the annunciator after an alarm and to set up for another alarm should one exist. A permanent plant modification to the alarm annunciation system is being planned.

The resident inspector was in the control room at the time of the event and has no further questions.

The NRC received reports of the above events via the ENS. The inspectors determined that the licensee's initial response and corrective actions were appropriate. The root cause analysis and the need for additional/long-term corrective action will be reviewed upon issuance of the Licensee Event Reports as part of the routine inspection program.

1.3 Engineered Safety Feature (ESF) System Walkdown (71710)

The inspectors verified the operability of the 'A' Loop of the Unit 2 Core Spray System by performing a walkdown of the system. The following procedures and drawings were used during the inspection:

S52.1.A	Core Spray Setup for Service Operatic
S52.1.A (COL-1)	Equipment Alignment for Core Spray Loop 'A' Operation
M-52	Core Spray System Piping and Instrumentation Drawing

During the walkdown, the inspectors confirmed that the system lineup and procedures agree with plant drawings and the as-built system configuration. The inspectors also looked for equipment conditions that may degrade system performance, verified that installed instrumentation was calibrated and functioning and valves were positioned and locked as appropriate. The inspectors found the systems to be properly aligned and in a good working condition. No concerns or problems were noted during this inspection.

2.0 SURVEILLANCE/SPECIAL TEST OBSERVATIONS (61726)

During this inspection period, the inspector reviewed in-progress surveillance testing and completed surveillance packages. The inspector verified that surveillances were done according to PECo approved procedures and plant Technical Specification requirements. The inspector also verified that the instruments used were within calibration tolerance and that qualified technicians did the surveillances.

Surveillance testing observed and/or reviewed included:

St-6-092-318-2 D24 Diesel Generator Fast Start Operability Test Run

This activity observed by the inspectors was acceptable.

3.0 MAINTENANCE OBSERVATIONS (62703)

The inspector reviewed the safety-related maintenance activities to verify that repairs were made in accordance with approved procedures and in compliance with NRC regulations and recognized codes and standards. The inspector also verified that the replacement parts and quality control used on the repairs were in compliance with PECo's Quality Assurance (QA) program. The following maintenance activity was reviewed:

CO 130857 1B Residual Heat Removal (RHR) System Heat Exchanger (Hx) Corrosion Monitoring Loop Installation (Modification 6221-1)

The activity observed by the inspectors was acceptable.

3.1 Motor Operated Valve Diagnostic Testing

On August 6, 1992, the inspector witnessed the performance of test procedure M-500-030, "Diagnostic Testing of Limitorque Motor Operated Valves (MOV)" on the outboard steam isolation valve, HV-49-2F008, for the Reactor Core Isolation Cooling (RCIC) System. This procedure provides instructions for diagnostic testing of Limitorque Motor Operators using the Valve Operation Test and Evaluation System (VOTES).

During the performance of the test procedure, the inspector noted that the torque switch setting (TSS) on the MOV did not meet the engineering pretest data. The minimum calculated TSS was 2.0 and the maximum calculated TSS was 2.25. The target thrust was 12,163 pounds and the limiting component thrust was 21,725 pounds for this actuator. The "as-found" TSS setting was 1.0 to open and 1.0 to close. A VOTES test was performed on the valve at the TSS of 1.0 and the results showed that the thrust at the torque switch trip (TST) was 4,209 pounds. The maintenance technicians recognized that the "as-found" thrust was not within the thrust window as specified in the engineering pretest data. The TSS was changed to the minimum calculated TSS of 2.0 and another VOTES test was performed. The test results showed that the TST was now within the thrust window. The thrust at the TST was now 16,615 pounds.

At the time of this test there were two maintenance technicians, a maintenance sub-foreman and a non-licensed operator present. The operator was required because the system was being tested while remaining operable. This was the first VOTES test performed while at power on an operable system.

When the TSS was found outside the minimum and maximum TSS, the valve was inoperable. The appropriate Limiting Condition for Operations (LCO) 3.6.3, as required by Technical Specifications (TS) was not entered. The PECo personnel performing the test followed the procedure successfully and were not aware that a LCO was entered. The personnel in the field were not TS trained and the procedure did not delineate actions to be taken if parameters were outside the design parameters listed within the test document. The valve was made operable after the TSS was adjusted. PECo was in this LCO for approximately 5 to 10 minutes. Because the time period was within 4 hours, allowed by the LCO, no violation of TS occurred.

The corrective action taken by maintenance engineering was a temporary change (TC) to the procedure. This was written on August 11, 1992. The TC incorporated notification of operations when "as-found" VOTES testing thrust values are less than or in excess of the design-basis values for administrative controls as dictated by technical specifications. This change was made permanent by the issuance of Revision 3 to M-500-030 dated August 28, 1992.

The inspector raised a concern with the maintenance engineering supervisor regarding HV-49-2F007, RCIC steam supply inboard isolation valve. This was an operability concern whether or not this valve would be capable of performing its function under design basis conditions based on the results of the VOTES testing performed on the outboard MOV. PECo maintenance engineering issued an engineering work request on August 11, 1992, for the Nuclear Engineering Department (NED) in Chesterbrook to perform a safety evaluation. NED evaluated test results for all four valves similar to HV-49-2F008. The "as-left" Motor Operated Valve Analysis and Test System (MOVATS) thrust at TST was 10,600 pounds. The Generic Letter (GL) 89-10 MOV calculation of record for the RCIC system assumes the worst case flow-to-seat orientation (i.e., flow under seat) with respect to valve closure thrust requirements. This valve has a flow-to-seat orientation that aids valve closure (i.e., flow over seat). A new calculation has been prepared to address the actual flow-to-seat orientation for operability assessment. This new calculation results in a minimum thrust to close of 7,255 pounds without error allowance. The Local Leak Rate Test (LLRT) history demonstrated sealing force at the existing TSS. The inspector reviewed the safety evaluation and concluded that it conforms to its intent.

During the Unit 1 Refuel Outage 4 (1RO4) 138 valves were VOTES tested. There were 15 instances were a MOV's "as-found" thrust did not achieve design-basis specification thrust as calculated by Chesterbrook engineering. These new calculations were performed based on MOVATS inaccuracies. The "as-found" thrust at the TST was less than the minimum required thrust and, therefore, a Non-Conformance Report (NCR) has been issued.

An operability/reportability assessment is being performed to determine if these valves could have performed their function under design basis conditions. Using the MOV's "as-found" thrust, Chesterbrook engineering will be analyzing actual valve failure scenarios. All 15 valves were adjusted during testing activities to above their design-basis specification thrust.

PECo is currently developing a controlled document that is called "MOV Integrated Data Acquisition System (MIDAS)." The MIDAS document contains two data bases. Data base A contains their record of design basis calculations, which is being updated based on industry standards and Nuclear Management and Resources Council (NUMARC) guidelines. The PECo Nuclear Engineering Service Department (NESD) is responsible for this data base. The expected date of completion for the updates is mid September 1992. Data base B contains the last testing data to be recorded, such as: TSS, stroke time, TST, total thrust and open torque switch bypass time. Currently this data base is being updated in accordance with the VOTES testing. The tests performed on the 138 valves tested during 1RO4 are being reviewed and the recorded data will be incorporated into MIDAS. The valves in Unit 2 that have been tested are now being entered into the MIDAS document as the tests are completed. When all of this data is entered into MIDAS it will be the living document for the MOV data base.

The inspector concluded that PECo is currently using a systematic program approach aided by NUMARC guidelines in updating present data and testing of MOVs. The inspector observed that there was a lack of communication between the maintenance engineering staff and the technicians performing the testing, however, the change to procedure M-500-030 should correct the communication weakness. Overall, the MOV program appears to be in adherence with the intent of GL 89-10.

4.0 ENGINEERING AND TECHNICAL SUPPORT (37700)

4.1 NRC Bulletin No. 92-01 Supplement 1: Failure of Thermo-Lag 330 Fire Barrier System to Perform Its Specified Fire Endurance Function

As a result of additional Thermo-Lag 330 fire endurance tests, Supplement 1 to NRC Bulletin No. 92-01 expanded the requested actions to licensees. Supplement 1 requested that licensees implement compensatory measures in all plant areas that have Thermo-Lag 330 fire barrier systems installed rather than in selected areas based on the particular Thermo-Lag 330 configuration.

As discussed in Section 4.1 of NRC Inspection Reports 50-352/92-17 and 50-353/92-17, PECo implemented compensatory measures in all plant areas using Thermo-LAG 330 fire barriers. Thus, PECo had already implemented the requested actions of Supplement 1 of NRC Bulletin 92-01 at the time the initial bulletin was issued. These actions exhibited a proactive response to the fire barrier issues.

4.2 Residual Heat Removal (RHR) System Corrosion Monitoring

The inspector reviewed plant modification 6221-1, which installs corrosion monitoring loops on both Unit 1 RHR heat exchangers. These loops are being added because of the corrosion found on the inside of the heat exchanger tubes during the last refueling outage.

The monitoring loop on the 1A heat exchanger is already installed and permits 12 sample tubes to be exposed to the same RHR service water as the 1A heat exchanger. The 1A heat exchanger is being maintained in a normal lay-up condition with demineralized water and will only be used as needed to perform surveillance tests or for normal operations when the 1B heat exchanger is unavailable. The corrosion monitoring loop is normally isolated from the heat exchanger by closed manual valves. Water from the heat exchanger is periodically circulated through the monitoring loop to ensure the conditions inside the sample tubes are representative of those inside the heat exchanger. The design of the monitoring loop permits the removal of a sample tube every two months for laboratory analysis. Since the 1A heat exchanger will be maintained in a lay-up condition, the design of the monitoring loop does not require the outside of the sample tubes be exposed to an environment which duplicates that within the heat exchanger.

The corrosion monitoring loop for the 1B heat exchanger is presently being installed and is of a different design from that used in the 1A heat exchanger. The 1B monitoring loop will be aligned to the heat exchanger continuously. RHR service water will flow though the 12 sample tubes whenever there is flow through the heat exchanger. A closed heating loop is also provided for in this design so that the outside of the tube is kept at the same temperature as the shell side of the RHR heat exchanger. As with the 1A design, one of the 12 sample tubes will be removed every two months for analysis.

The use of these monitoring loops is a good initiative and should allow PECo to track the condition of the heat exchanger tubes during the current operating cycle.

5.0 RADIOLOGICAL PROTECTION (71707)

During the report period, the inspector examined work in progress in both units including health physics procedures and controls, As Low As Reasonably Achievable (ALARA) implementation, dosimetry and badging, protective clothing use, adherence to radiation work permit (RWP) requirements, radiation surveys, radiation protection instrument use, and handling of potentially contaminated equipment and materials.

5.1 Contamination Control Problems on Site

Two recent incidents connected with contamination controls suggest that some weaknesses may exist in this area of PECo's radiological controls program. The first incident occurred on August 6, 1992, at about 5:00 p.m. A worker was leaving the protected area through the portal monitors located at the security area of the Administration Building. The person alarmed the monitor and immediately left the building, and was not stopped by the security guards on duty at the time. The security officers are located in the general area of the portal monitors but not in direct view of them. However, the alarms are clearly audible at the guard's location. Leaving site following a portal monitor alarm is contrary to proper radiological practice. Two NRC inspectors who happened to be in the area at the time alerted security to this event. The person involved had by that time left site, but a search through the security badges, collected at that time, enabled the NRC inspectors to tentatively identify the person. That person was contacted at his residence and asked to return immediately to the site in his work clothes. The person returned a few hours later and was checked for contamination, but none was found. A PECo representative stated that the individual had not worked in contaminated areas that day.

The guidance for exiting through the portal monitors, instructions on the use of the monitors, and the actions to take for an alarm, are all discussed during the initial General Employee Training (GET) provided to all new plant employees. A video tape on the use of the monitors and the nature of the alarms is also shown during the training. Following an ala m, correct practice requires that the person recount on the monitor and, if no alarm is rece ved a second time, the person may leave the site. However, if a second alarm is received, the person should contact Health Physics for assistance.

Discu sions with security personnel and a review of security Post Order No. 3, "Exit Control - Administration/Technical Support Center Buildings," revealed that observing the use of the portal monitors and ensuring proper response to alarms was not part of the security officer's duties. The security officer is required to observe portal monitor operations only during actual or training emergency conditions. The security officer's training material provides guidance on the utilization of the portal monitors and the actions to take for an alarm, but it does not mention that these actions are limited only to emergency situations, and the material, therefore, is inconsistent with the Post Order. In addition, the training material instructs the security officers to contact Health Physics via security shift supervision if anyone exits without clearing the monitors, again in apparent conflict with the Post Order. Security personnel stated that the security officers will respond to portal monitor alarms, but only during off-peak traffic times. They also stated that during high traffic times, the security officers are busy monitoring personnel keying out of the protected area and collecting the security badges, and would not be able to simultaneously react to portal monitor alarms. PECo representatives stated that the portal monitors frequently alarm for reasons other than the presence of contamination, primarily because of improper breaking of the personnel-sensing beam located in the monitors and used to start the count when a person steps into the monitor. PECo representatives stated that improper motion through the monitor, or improper carrying of an article such as a bag through the monitor, will sometimes cause an alarm.

The portal monitors at the security point are provided as an additional contamination control measure and are not required by 10 CFR 20. The primary contamination controls monitors are located at the exits from contaminated areas, also at the exits from the radiological controls areas (RCA). All personnel leaving the protected area must pass through the portal monitors at the security point. Persons leaving the RCA must pass through whole body friskers at the RCA exits. With few exceptions, the RCA is located within the protected area.

PECo representatives stated that they will review the use of the portal monitors and will take appropriate corrective actions to ensure proper use of these monitors. This item will be reviewed during future inspections.

The second incident involving improper contamination controls practices occurred on the day following the incident described above. During a routine surveillance of the instrument shops located outside the RCA, PECo technicians discovered survey instruments with contamination levels above those allowed in these areas by station procedures. The contamination was mostly fixed, but some smearable contamination was also found. The survey instruments had been checked for contamination by HP technicians before leaving the

RCA, and the licensee believed that the instruments had been brought out of the RCA over an extended period of time and involved contamination surveys performed by several HP technicians. Articles brought out of the RCA are considered released for unrestricted use and no further surveys are required to be performed on them. PECo representatives stated that they did not have all the details connected with these contaminated instruments but that an investigation had been initiated, and corrective actions will be taken based on the findings of the investigation. This item will, therefore, be reviewed during a future inspection.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

6.1 Closure of Region I Temporary Instruction Regarding Falsification of Records

The inspectors made several plant tours in all areas of the plant as per directions within the Temporary Instruction. One tour was made on deep back shift. The inspectors concluded that the shift personnel interviewed knew what their job entailed and knew the importance of taking readings assigned to that position. The inspectors also concluded that the accompanied shift personnel were proficient in the performance of their duties.

6.2 Reactor Water Level Instrumentation

During this report period an issue regarding reactor vessel water level instrumentation in Boiling Water Reactor's (BWR's) was identified at Northeast Utilities Millstone Unit 3. The problem is the inaccuracies of water level indication during and after a rapid depressurization event could cause dissolved non-condensable gasses to come out of solution. This could cause a level decrease in the reference leg of the level detector, resulting in a false high indication on the vessel level instruments.

The NRC responded by issuing an Information Notice (IN) 92-54 dated July 24, 1992 and a GL 92-04 dated August 19, 1992. Between the IN and the GL the NRC staff held a public meeting with the Regulatory Response Group (RRG) of the Boiling Water Reactor Owners Group (BWROG) to discuss the effects of the inaccuracies of the water level indication. The above documents are in the public record.

On August 17, 1992, PECo briefed the Senior Resident Inspector on PECo's position on the water level issue and Limerick's progress to date. Attached in Attachments 1 and 2 are the handouts from the briefing. Attachment 1 delineates the chronology of the issue, the short and long term action plan, and a technical discussion regarding water level instrumentation. Attachment 2 discusses the BWROG plan and schedule for the actions regarding reactor vessel level instrumentation. Limerick has committed to follow the BWROG plan.

To date, the following have been confirmed by the resident inspector:

1. PECo has responded to BWROG Vessel Level Instrumentation Survey.
2. PECo does not use Yarway instruments. (Reference GL 83-24)

3. PECo has not experienced any level spiking during a depressurization event below 400 pounds.
4. PECo has conduced special training to alert all operators to the level spiking issue. (Attachment 3)
5. Limerick's vessel level instrumentation is installed per GE Service Information Letter SIL-470.
6. All level transmitters are Rosemont.
7 The level instrumentation is calibrated wet and "head chambers" are used at the instrument racks to minimize air intrusion into the reference logs.

The resident inspectors will continue to follow PECo's progress regarding reactor water level instrumentation.

7.0 REVIEW OF LICENSEE EVENT REPORTS (LERs), ROUTINE AND SPECIAL REPORTS (90712, 92700)

7.1 Licensee Event Reports (LERs)

LERs are 30 day reports submitted to the NRC, by PECo, as required by 10 CFR 50.73. These reports document: the major occurrences present during an event, including all component or system failures; a clear, specific, narrative description of what occurred; plant operating conditions before the event; status of contributors to the event; dates and approximate times of contributing factors; the causes and failure modes; personnel errors if applicable; procedural deficiencies if applicable and the short-term and long-term corrective actions taken to prevent recurrence. The Resident Inspector routinely reviews these documents and performs follow-up to PECo's actions regarding the disposition of corrective initiatives. In his review, the inspector validates the above and determines whether events are described accurately and whether corrective and compensatory actions have been properly addressed. Unless otherwise delineated below, the following LERs meet all the requirements discussed above.

<u>LER 1-92-013, Event Date: June 24, 1992, Report Date: July 20, 1992</u>
<u>Inadequate Surveillance Test</u>

As discussed in Section 3.3 of NRC Inspection Report 50-352/92-17 and 50-353/92-17 PECo had not been verifying a flow rate of 10,000 gallons per minute (gpm) through the RHR Hx as required by TS 4.6.2.3.b. Instead, the surveillance procedure measured total loop flow that was the sum of the flow through the RHR Hx and any leakage past the associated Hx bypass valve. In some cases, the bypass valve flow was significant enough to result in less than 10,000 gpm flow through the Hx although the loop flow exceeded 10,000 gpm.

A PECo analysis of the test data and plant operating conditions determined that the RHR Hx's on both Units 1 and 2 would have been able to remove the design heat loads from the suppression pool.

Upon identification of the problem PECo removed flow restricting orifice plates in the flow paths that resulted in total loop flow being increased to a point that assured the TS required flow through the Hx was being satisfied. The surveillance tests were revised such that Hx flow and not only total loop flow rates are verified.

A TS change has been submitted to the NRC to clarify TS 4.6.2.3.b as to what the required flow and flow path should be. Based on the corrective actions taken the inspector had no further questions and unresolved items 50-352/92-27-01 and 50-353/92-17-01 are closed

LER 1-92-014, Event Date: June 26, 1992, Report Date: July 24, 1992
Inoperable Fire Rated Barrier

During a review of fire barrier installations, PECo discovered that an inoperable fire barrier in the control enclosure was being monitored by an hourly fire watch patrol when a further review of the circumstances indicated a continuous fire watch was required by plant TSs. A continuous fire watch was immediately posted and station fire protection personnel have been apprised of the occurrence and the unique areas of the plant that may be susceptible to such a misinterpretation.

LER 1-92-015, Event Date: July 7, 1992, Report Date: August 6, 1992
High Pressure Coolant Injection (HPCI) System Failure

This LER reported the failure of the HPCI system to perform properly during surveillance testing following maintenance. The failure occurred when particles clogged equipment associated with the hydraulic-mechanical overspeed trip mechanism. The source of the particles could not be identified, however, the HPCI maintenance procedure are being revised to ensure the adequate inspections and flushes of the systems are performed to preclude recurrence.

LER 2-92-007, Event Date: June 24, 1992, Report Date: July 17, 1992
Improper Restoration of Ventilation

This LER reported a TS violation that occurred when a reactor operator failed to properly reset a Unit 2 reactor enclosure secondary containment isolation. The operator failed to perform a step of the system operating procedure resulting in the reactor enclosure secondary containment low differential pressure isolation being inadvertently bypassed for approximately four hours. The operator discovered his error during a panel walkdown and then returned the affected switches to their normal positions.

The inspector found the corrective actions taken in response to this event to be comprehensive and had no further questions regarding this event.

LER 2-92-008, Event Date: July 17, 1992, Report Date: August 12, 1992
Reactor Enclosure Partial Isolation

This LER reported the actuation of various Primary Containment and Reactor Vessel Isolation Control systems and a Unit 2 Reactor Enclosure Secondary Containment Isolation. The cause of the isolations was a blown fuse in the isolation logic circuitry. The cause of the blow fuse could not immediately be determined and the fuse has been sent to the manufacturer for failure analysis. The blown fuse was replaced and all isolations were reset within 41 minutes with no adverse effects on the plant.

LER 2-92-009, Event Date: July 28, 1992, Report Date: August 25, 1992
Primary Containment and Reactor Vessel Isolation Actuation

Refer to Section 1.2 of this report for details of this event.

7.2 Routine and Special Reports

Routine and special reports are submitted by PECo to inform the NRC of routine operating conditions and other noteworthy occurrences that are reportable due to requirements in 10 CFR 20, technical specifications and other regulatory documents. The inspector reviews these reports for information and confirms the accuracy of the reports. The following report was reviewed and unless otherwise delineated below, satisfied the requirements for which it was reported.

Monthly Operating Report for July 1992, dated August 10, 1992

The resident inspector had no further concerns or questions regarding the above listed report.

8.0 FOLLOWUP OF PREVIOUS INSPECTION FINDINGS (92702)

(Closed) 50-352/91-16/02 Inoperable Off-Site Power Source

The inspector reviewed LER 1-91-017 and Supplement 1 to the subject LER and concluded that PECo has adequately evaluated the effects of a missing fuse in the automatic voltage controller for the 201 Safeguards Transformer. The inspector had no further questions regarding this event. This item is considered closed.

(Closed) Unresolved Item Nos. 50-352/92-17-01 and 50-353/92-17-01.

Based on the corrective actions taken by PECo to resolve RHR testing questions these items are closed. Refer to the review of LER 1-92-013 (Section 7.1 of this report) for additional details.

9.0 MANAGEMENT MEETINGS

9.1 Exit Interviews

The NRC Resident Inspectors discussed the issues in this report with PECo representatives throughout the inspection period, and summarized the findings at an exit meeting with the Plant Manager, Mr. J. Doering, on September 2, 1992. No written inspection material was provided to licensee representatives during the inspection period.

9.2 Additional NRC Inspections this Period

The Resident Inspector also attended the following exit interviews during the report period:

Date	Inspector	Report	Subject
July 29, 1992	S. Hansell	50-352/92-21 50-353/92-21	License Examinations

CHRONOLOGY OF ISSUE

FEBRUARY 1989 -	WESTINGHOUSE ELECTRIC CORPORATION ISSUES NOTICE TO PWR OWNERS OF POSSIBLE PRESSURIZER LEVEL INDICATION ERROR DURING RAPID DEPRESSURIZATION DUE TO NONCONDENSIBLE SATURATION OF COLD LEG LEVEL INSTRUMENTATION'S REFERENCE LEG
FEBRUARY 1992 -	NRC REQUESTS BWR OWNERS GROUP (BWROG) TO ADDRESS POSSIBLE REACTOR WATER LEVEL INDICATION ERROR FOLLOWING A RAPID DEPRESSURIZATION AS A RESULT OF NONCONDENSIBLE SATURATION OF REFERENCE LEG
JULY 21, 1992 -	NORTHEAST UTILITIES DECLARES REACTOR WATER LEVEL INDICATIONS (FROM COLD LEG INSTRUMENTS) INOPERABLE FOR POST-ACCIDENT MONITORING DUE TO NONCONDENSIBLE SATURATION OF REFERENCE LEGS
JULY 22, 1992 -	NRC ACTIVATES BWROG REGULATORY RESPONSE GROUP (RRG)
JULY 24, 1992 -	NRC ISSUES NRC INFORMATION NOTICE 92-54
JULY 27, 1992 -	BWROG RRG MEETS TO REVIEW ISSUE
JULY 29, 1992 -	BWROG MEETS WITH NRC TO DISCUSS ISSUE
AUGUST 12, 1992 -	BWROG SUBMITS SCHEDULE TO NRC FOR RESPONSE TO OPEN ITEMS FROM JULY 29 MEETING AND PROPOSES LONG TERM PROGRAM TO EVALUATE PHENOMENA AND ITS EFFECTS

ACTION PLAN

SHORT TERM

BWROG

* AUGUST 28, 1992 -

  SUBMIT GENERIC REPORT TO THE NRC, INCLUDING RESOLUTION OF "SHORT TERM OPEN ITEMS". ENSURE ISSUANCE OF REVISION TO SIL 470.

* EPC COMMITTEE TO CONSIDER ISSUANCE OF SPECIFIC DIRECTION TO OPERATIONS PERSONNEL FOR POST-ACCIDENT OPERATIONS

UTILITIES

* EVALUATE NEED TO MAKE OPERABILITY DETERMINATION

* SENSITIZE OPERATIONS PERSONNEL TO ISSUE AND POSSIBLE ASSOCIATED REACTOR WATER LEVEL INDICATION ERROR

* SEPTEMBER 29, 1992 - UTILITIES TO RESPOND IN ACCORDANCE WITH NRC GENERIC LETTER (TO BE ISSUED)

ACTION PLAN

LONG TERM

BWROG

* LONG TERM PROGRAM

  - PERFORM CONTROLLED TEST FOR EFFECTS OF RAPID DEPRESSURIZATION ON REFERENCE LEG CONFIGURATIONS UNDER NONCONDENSIBLE GAS SATURATION CONDITIONS

  - PERFORM CONTROLLED TEST ON CONDENSING CHAMBER CONFIGURATIONS TO DETERMINE CRITICAL CHARACTERISTICS OF PERFORMANCE AND QUANTIFY NONCONDENSIBLE GAS CONCENTRATIONS

  - ESTABLISH ANALYTICAL MODEL FOR DETERMINING DEPRESSURIZATION AFFECTS ON PLANT SPECIFIC PIPING CONFIGURATIONS

  - DETERMINE VALUES OF ACCEPTABLE REACTOR WATER LEVEL ERROR

  - REVIEW POSSIBLE MODIFICATIONS TO PLANT SYSTEMS OR PROCEDURES TO COMPENSATE FOR LEVEL INDICATION ERRORS, IF REQUIRED.

  - INVESTIGATE METHODS OF OBTAINING REFERENCE LEG INVENTORY SAMPLES

UTILITIES

* EVALUATE NEED TO MAKE OPERABILITY DETERMINATION

* IF REQUIRED, IMPLEMENT MODIFICATIONS TO PLANT SYSTEMS OR PROCEDURES TO ACCOMMODATE REACTOR WATER LEVEL INDICATION ERROR

TECHNICAL DISCUSSION

ISSUE:

1. CAN THE REFERENCE LEGS OF REACTOR WATER LEVEL COLD LEG INSTRUMENT SYSTEMS BECOME SATURATED WITH NONCONDENSIBLE GASES?

2. WHAT IS THE EFFECT OF A REACTOR RAPID DEPRESSURIZATION ON A SATURATED REFERENCE LEG?

3. WHAT IS THE SAFETY SIGNIFICANCE OF ANY RESULTANT LEVEL ERROR INDICATION?

WHAT WE KNOW TO DATE:

I. ANALYTICAL EVALUATION (TO DATE)

TO DATE, NO PHYSICAL TESTING HAS BEEN PERFORMED TO BOUND THIS ISSUE. IN ORDER TO EVALUATE THE IMPACT OF POSSIBLE REACTOR WATER LEVEL INDICATION ERROR, WORST CASE PARAMETERS AND CONDITIONS HAVE BEEN ASSUMED IN BOUNDING CALCULATIONS AND MODELS.

1. CAN THE REFERENCE LEGS OF REACTOR WATER LEVEL COLD LEG INSTRUMENT SYSTEMS BECOME SATURATED WITH NONCONDENSIBLE GASES?

ANALYSES HAVE SHOWN THAT THE REFERENCE LEGS CAN BECOME SATURATED WITH NONCONDENSIBLE GASES. THE TWO IDENTIFIED MECHANISMS FOR SATURATION ARE: DIFFUSION AND MASS TRANSPORT THROUGH SYSTEM SEEPAGE/LEAKAGE. RELATIVELY, DIFFUSION WOULD BE A SLOW PROCESS IN COMPARISON TO SYSTEM SEEPAGE/LEAKAGE.

THE CONCENTRATION OF THE NONCONDENSIBLE GASES IN THE REFERENCE LEG IS DIRECTLY RELATED TO THE EQUILIBRIUM CONCENTRATION OF NONCONDENSIBLES IN THE ASSOCIATED CONDENSING CHAMBER. THE EQUILIBRIUM CONCENTRATION OF NONCONDENSIBLES IN THE CONDENSING CHAMBERS IS FUNCTION OF THE NORMAL CONCENTRATION OF NONCONDENSIBLES IN REACTOR STEAM, SYSTEM PRESSURE, AND SYSTEM TEMPERATURE. IT IS BELIEVED THAT THE EQUILIBRIUM CONCENTRATION OF NONCONDENSIBLE GASES IS ALSO A FUNCTION OF THE PLANT SPECIFIC CONFIGURATION OF THE CONDENSING CHAMBER AND ASSOCIATED PIPING. IN ADDITION, SYSTEM LEAKAGE MAY INCREASE THE CHAMBER EQUILIBRIUM CONCENTRATION OF NONCONDENSIBLES DEPENDENT ON THE LEAK SIZE. THE RESULTANT CONCENTRATION OF NONCONDENSIBLE GASES IN THE REFERENCE LEG IS ALSO A FUNCTION OF SYSTEM PRESSURE AND TEMPERATURE.

BASED ON A STEAM CONCENTRATION OF 25PPM H2 AND 5PPM O2 (BY VOLUME), THE EQUILIBRIUM CONCENTRATION OF NONCONDENSIBLE GASES IN A PROPERLY FUNCTIONING CONDENSING CHAMBER HAS BEEN EVALUATED TO BE APPROXIMATELY 3 - 4% (30 - 40 PSI PARTIAL PRESSURE).

2. WHAT IS THE EFFECT OF A REACTOR RAPID DEPRESSURIZATION ON A SATURATED REFERENCE LEG?

ANALYSES DONE TO DATE HAVE BEEN BASED ON THE VOLUMETRIC DISPLACEMENT OF WATER BY HYDROGEN IN AN OPEN AND STRAIGHT VERTICAL PIPE WITH A LENGTH OF TEN FEET. ONE SUCH ANALYSIS ASSUMED INSTANTANEOUS DEPRESSURIZATION (985PSIG - 2PSIG) AND WAS PERFORMED FOR NONCONDENSIBLE CONCENTRATIONS AT PARTIAL PRESSURES OF 30 PSIG AND 1000 PSIG. THE RESULTS SHOWED THE LOSS OF INVENTORY TO BE:

FOR 30 PSIG H2 :	3% LOSS OF INVENTORY, OR 4 LINEAR INCHES
FOR 1000 PSIG H2:	100% LOSS OF INVENTORY

A SECOND ANALYSIS WAS PERFORMED TO EVALUATE AT WHAT PRESSURES VOLUMETRIC DISPLACEMENT TAKES PLACE DURING THE INSTANTANEOUS DEPRESSURIZATION AND WHAT THE MAGNITUDE OF DISPLACEMENT WOULD BE IN THE SAME TEN FOOT PIPE. THIS SECOND ANALYSIS REVEALED THE FOLLOWING DISPLACEMENTS (LOSS OF LINEAR INVENTORY) FOR NONCONDENSIBLE CONCENTRATIONS OF 1000 PSIA AND 100 PSIA (ASSUMED TO BE H2):

	1000 PSIA	100 PSIA
@1000 PSIA	0 FT	0 FT
@ 400 PSIA	0.4 FT	0.04 FT
@ 100 PSIA	2.2 FT	0.22 FT
@ 50 PSIA	4.7 FT	0.47 FT
@ 17 PSIA	10 FT	1.48 FT

IT IS CONCLUDED THAT ANY POSTULATED REACTOR WATER LEVEL INDICATION ERROR SHOULD NOT BE OBSERVED UNTIL DEPRESSURIZATION BELOW APPROXIMATELY 450 PSIG. UPON DEPRESSURIZATION BELOW THAT POINT, THE INDICATED ERROR WOULD INCREASE.

THE ANALYSES TO DATE HAVE NOT TAKEN INTO ACCOUNT PIPE LENGTHS AT VARIOUS TEMPERATURES, OF VARIOUS MATERIALS, CONSISTING OF VARIOUS GEOMETRIES, OR WITH VARIOUS RESTRICTIONS. THE ANALYSES ALSO HAVE NOT EVALUATED VARIOUS RATES OF DEPRESSURIZATION. IN ORDER TO FURTHER EVALUATE AND BOUND THE POSSIBLE DEPRESSURIZATION CONCERN, THE BWROG HAS OUTLINED A LONG TERM PROGRAM WHICH INCLUDES CONTROLLED PHYSICAL TESTS AND THE CREATION OF AN ANALYTICAL MODEL TO SIMULATE ACTUAL PLANT SYSTEM PARAMETERS.

3. WHAT IS THE SAFETY SIGNIFICANCE OF ANY RESULTANT LEVEL ERROR INDICATION?

ON JULY 29, 1992, THE BWROG MET WITH THE NRC TO DISCUSS THE ISSUE OF NONCONDENSIBLE SATURATION OF THE REFERENCE LEG ON REACTOR WATER LEVEL COLD LEG INSTRUMENT SYSTEMS AND THE EFFECTS OF RAPID DEPRESSURIZATION. BASED ON THE ANALYSES DISCUSSED IN #2 ABOVE, GENERAL ELECTRIC DETERMINED THAT "DESIGN BASIS ANALYSES" WOULD NOT BE AFFECTED BY THE POSTULATED REACTOR WATER LEVEL INDICATION ERRORS. EVALUATION OF THE ANALYSES SHOWED THAT AUTOMATIC SAFETY SYSTEM INITIATIONS OCCUR BEFORE RPV DEPRESSURIZATION COULD INDUCE SIGNIFICANT WATER LEVEL ERRORS. IN ADDITION, SAFETY SYSTEMS INITIATIONS BY HIGH DRYWELL PRESSURE ARE UNAFFECTED.

WHILE AUTOMATIC SAFETY SYSTEM INITIATIONS ARE ENSURED, POST-ACCIDENT MONITORING MAY BE IMPACTED BY THIS PHENOMENA. THE BWROG DETERMINED THAT THE EMERGENCY PROCEDURE GUIDELINES HAVE BEEN SHOWN TO ADDRESS THE DESIGN BASIS EVENTS. EVENTS BEYOND THE DESIGN BASIS WILL HAVE TO BE EVALUATED FOR EFFECTS OF THE PHENOMENA, IF SHOWN PROBABLE. IN VIEW OF THIS, THE BWROG'S LONG TERM PROGRAM WILL EVALUATE THE PROBABILITY OF THIS PHENOMENA. IF RAPID DEPRESSURIZATION IS SHOWN TO CREATE REACTOR WATER LEVEL ERRORS DUE TO NONCONDENSIBLE SATURATION OF THE REFERENCE LEGS, THE BWROG PLANS TO DETERMINE A METHOD OF QUANTIFYING THE ERROR AND PROVIDE GUIDANCE TO EACH UTILITIES ON HOW TO EVALUATE THE PROJECTED LEVEL INDICATION ERROR.

II. RECENT INDUSTRY OBSERVATIONS

1. BOSTON EDISON - PILGRIM

DURING NORMAL COOLDOWN/DEPRESSURIZATION, FLUCTUATIONS IN REACTOR WATER LEVEL INDICATION HAVE BEEN OBSERVED BELOW 400PSIG. THE MAGNITUDE OF FLUCTUATIONS ARE GREATER ON ONE OF THE REACTOR WATER LEVEL INSTRUMENT REFERENCE LEGS. THE "B" REFERENCE LEG FLUCTUATIONS ARE A MAXIMUM OF +4" AT 450PSIG, BUT INCREASE TO +20" AT < 100PSIG. THE "A" REFERENCE LEG REMAINS STABLE UNTIL 65PSIG. THE FLUCTUATIONS ARE CYCLIC WITH INDICATED LEVEL ALWAYS RETURNING TO NORMAL. THESE FLUCTUATIONS HAVE BEEN OBSERVED SEVERAL TIMES OVER THE PAST 2.5 YEARS.

2. NORTHEAST UTILITIES

DURING THE LAST NORMAL COOLDOWN/DEPRESSURIZATION, FLUCTUATIONS IN REACTOR WATER LEVEL INDICATION WERE OBSERVED BELOW 450PSIA. THE FLUCTUATIONS WERE CYCLIC AND REACHED A MAXIMUM OF +2" INDICATED REACTOR WATER LEVEL.

BWR OWNERS' GROUP

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BWROG-92072
August 12, 1992

Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Mail Station 12 G18
Washington, DC 20555

Attention: Mr. William T. Russell, Associate Director
Inspection & Technical Assessment

Subject: REACTOR VESSEL WATER LEVEL INSTRUMENTATION

Enclosures:
1) Draft Outline of Generic BWROG Report Revision 1
2) BWROG Reactor Water Level Instrumentation Long Term Action Plan
3) Utility Commitment List

Reference: Letter, G. J. Beck (BWROG/RRG) to W. T. Russell (NRC), "BWR Owners' Group Transmittal Re: Action Items on Reactor Vessel Water Level Instrumentation for Boiling Water Reactors", August 5, 1992

This letter presents the BWR Owners' Group (BWROG) plan and schedule for actions to address postulated errors in BWR water level measurement instrumentation due to non-condensable gas in the reference columns. These actions were outlined in the reference letter.

Responses to the action items identified as short term in the reference letter will be addressed in the revised generic report to be submitted August 28, 1992. In response to requests from the NRC staff, action items 6, 9, 12 and 28, which were previously identified as long term, are now expected to be addressed as short term. Because of the near term completion date and the specificity of the individual items, detailed planning and scheduling information is not necessary for the short term items. Enclosure 1 outlines the revised generic report.

The BWROG has formulated a comprehensive long term action plan described in Enclosure 2 which embodies plant unique tests, analysis, and modifications if necessary. This plan will resolve the issues of concern as well as address action items 7, 8, 11, 13, 14, 16, 17, 18, 19 and 20.

The approach being taken is primarily based on full scale tests to measure reference leg inventory depletion expected during prototypical depressurization with prototypical concentrations of non-condensables. The plan includes the development, validation and use of an analytical model for better understanding of the phenomena, and plant-specific calculations if it should be found that they are required. The program also includes identification and evaluation of modifications which may be implemented should the prototypical full scale tests and analysis indicate unacceptable level measurement instrumentation error.

This action plan was developed by a team of technical experts from GE, EPRI and several utilities who are familiar with the issues of concern as well as the phenomena and analytical techniques involved. In addition, consideration is being given to available literature, academic resources and peer review.

This team, under the guidance of the BWROG, is dedicated to the technically appropriate, expeditious resolution of this issue.

Estimated completion dates for the tasks included in the action plan are shown in Figure 1 of Enclosure 2. To the extent practical, activities are being conducted in parallel. We are currently developing detailed responsibility assignments, cost estimates, and schedules based on this action plan.

Completion of action item 5 by individual utilities is contingent upon action item 15 as well as information from the long term action plan.

The BWROG has informed its member utilities of the NRC's expectation that each utility will provide to the NRC a response regarding the applicability of the BWROG generic report to each of the utility's plants, within 30 days of their receipt of the generic report. The utilities listed in Enclosure 3 communicated by noon August 12 a commitment to make their response within 30 days.

Very truly yours,

George J Beck

George J. Beck, Chairman
Regulatory Response Group
BWR Owners' Group

Enclosures

cc: T. E. Collins, NRC
S. F. Newberry, NRC
A. C. Thadani, NRC
BWROG Executives
BWROG Regulatory Response Group
BWROG Primary Representatives
C. L. Tully, BWROG Chairperson
L. A. England, BWROG Vice Chairman
B. T. Williamson, EPC II Chairman
J. E. Dale, GE
S. J. Stark, GE
R. C. Torok, EPRI
G. Oakley, INPO
T. P. Matthews, NUMARC
NRC Public Document Room
BWROG Technical & Licensing Contacts

Enclosure 1

DRAFT OUTLINE OF GENERIC BWROG REPORT REVISION 1

- o Introduction/Purpose
    - Address original NRC question
    - Respond to additional NRC questions from July 29 meeting
- o Conclusions
    - Safety significance
    - Further actions
- o Plant Configurations of the Water Level Instruments
    - Ideal configuration
    - Typical plant configurations
- o "Cold Leg" Instrumentation
    - Description and design basis
    - Performance
    - Potential problems that could disable the instrumentation
- o Analysis of Gases Coming Out of Solution
- o Discussion of Reference Leg Under Depressurization Conditions
- o Safety Analysis
    - Protection system and operator responses
        - LOCA inside and outside drywell
        - ADS
        - Other events
    - Benchmarking of codes against event data
    - Other instruments for safety system initiation
    - Emergency Procedure Guidelines
    - Operator response

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Enclosure 2

BWROG REACTOR WATER LEVEL INSTRUMENTATION LONG TERM ACTION PLAN

Description

General

Figure 1 schematically describes the entire program. It embodies both experimental and analytical bases in redundant fashion. The principal success path is the demonstration by prototypical testing that in-plant reference leg configurations operate acceptably with dissolved non-condensables. In parallel with this, an analytical reference leg model will be developed. The model will enhance physical understanding and will permit plant-specific calculations in cases where the test data may not be sufficiently applicable to a given plant configuration. If neither the test data nor plant unique calculations confirms acceptable operation, reference leg temperature and dissolved gas concentration data may be used to verify satisfactory operation. If none of these approaches succeed, a hardware and/or procedural modification is indicated. The plan includes a task to identify and fully evaluate a spectrum of modification alternatives.

Plant-Specific Data

This task consists of a comprehensive compilation of plant-specific condensing chamber and reference leg geometry and water chemistry data. This data will be used in bounding the variables for the Reference Leg De-Gas Test and the Condensing Chamber Performance Test.

Full Scale Reference Leg De-Gas Tests

The amount of inventory lost from de-gassing of the cold reference leg during depressurization is strongly dependent upon leg geometry, amount of initial non-condensable gases in the leg and the depressurization rate. This test will establish the amount of reference leg water which will be displaced during depressurization through full scale testing on prototypical cold leg piping geometries. Refer to Figure 2.

Preliminary test planning envisions the following: Prototypical mock-ups will be constructed which will permit the simultaneous depressurization of several cold leg piping geometries. The RPV will be simulated by a large vessel which is initially filled with N2. The tests will be initialized by backfilling the reference legs with water. The reference leg will then be saturated by slowly bubbling H2 through the leg. The amount of gas dissolved in the leg will be controlled by vessel pressure during this bubbling period. Once the leg is saturated the test will proceed by raising the large vessel pressure to 1000 psia and then by opening the discharge valve on top of the large vessel in a prescribed manner to achieve target depressurization rates. Upon completion of the depressurization the water remaining in the leg will be measured. Retention of the displaced water in the condensing chamber will be simulated. The test program will measure cold reference leg inventory lost as a function of initial non-condensable gas concentration in the cold leg, depressurization rate, and reference leg geometry.

Full Scale Condensing Chamber Performance Tests

Data to determine the performance of condensing chambers (CCs) will be obtained from a full scale test prototypical mock-up shown schematically in Figure 3.

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The following description of the test is based on planning to date: Because the test program must be run for a sufficient period to develop reliable data, several CC and inlet piping geometries will be tested simultaneously. Inlet piping length and slope as well as condensing chamber size and orientation will be simulated. The primary data from this test will be the non-condensable concentration entering the cold leg. This will be determined by periodically drawing out and analyzing a sample of water from the bottom of the CC. The CC will be instrumented with thermocouples to correlate temperature with CC performance, and small samples of gas may periodically be drawn from the top of the CC for subsequent analysis. Non-condensable gas in the vessel simulating the RPV (steam source) will be controlled to prototypical values during the test program. The test program will obtain at full scale the amount of non-condensable gas entering the reference leg from the CC as a function of CC inlet piping geometry, CC geometry, CC upper and lower external surface temperature and non-condensable gas in the upper region of the CC.

Test Data Applicability to Plant

The test programs will simulate a practical range of geometries. On completion, the applicability of the data to each plant will be evaluated. Qualifying plants will not need plant-specific calculations.

Analytical Model

This task will develop a transient analytical model which incorporates basic thermal-hydraulic phenomena, and two-phase flow effects, to predict the indicated level error for a given vessel depressurization. For a specific reference leg and condensing pot geometry, the error in the indicated level will be calculated and the results will be compared with the data from tests for a range of parameters.

Model Validation

Validation of the model against the reference leg de-gas tests will justify its use for plant-specific calculations.

Plant-Specific Calculations

If necessary, the validated analytical model will be configured to represent a given plant-specific geometry. Case runs will be made to determine the reference cold leg level variation under various operating conditions.

Acceptance Criteria Identification

Based on the postulation that the tests and analyses will predict some amount of instrument error, an acceptable error range will be determined for each class of BWR considering design and operational requirements.

Level Error Acceptability

The test results or plant-specific calculations will be compared to the acceptance criteria. The comparison will determine the need for plant or procedure modifications.

Reference Leg Samples / Condensing Chamber Temperature

If the test data cannot be shown to be applicable to a given plant and if the analytical model cannot be shown to be valid for plant-specific calculations, plant-specific data will be gathered. Procedures will be established for utility sampling of reference leg non-condensable gas and condensing chamber temperatures. These plant unique data, taken with reference leg de-gas tests and full scale condensing chamber performance test data, will allow utilities on a plant unique basis to assess the acceptability of the non-condensable gas concentrations which exist in their plant's reference legs.

Non-Condensable Concentration Acceptability

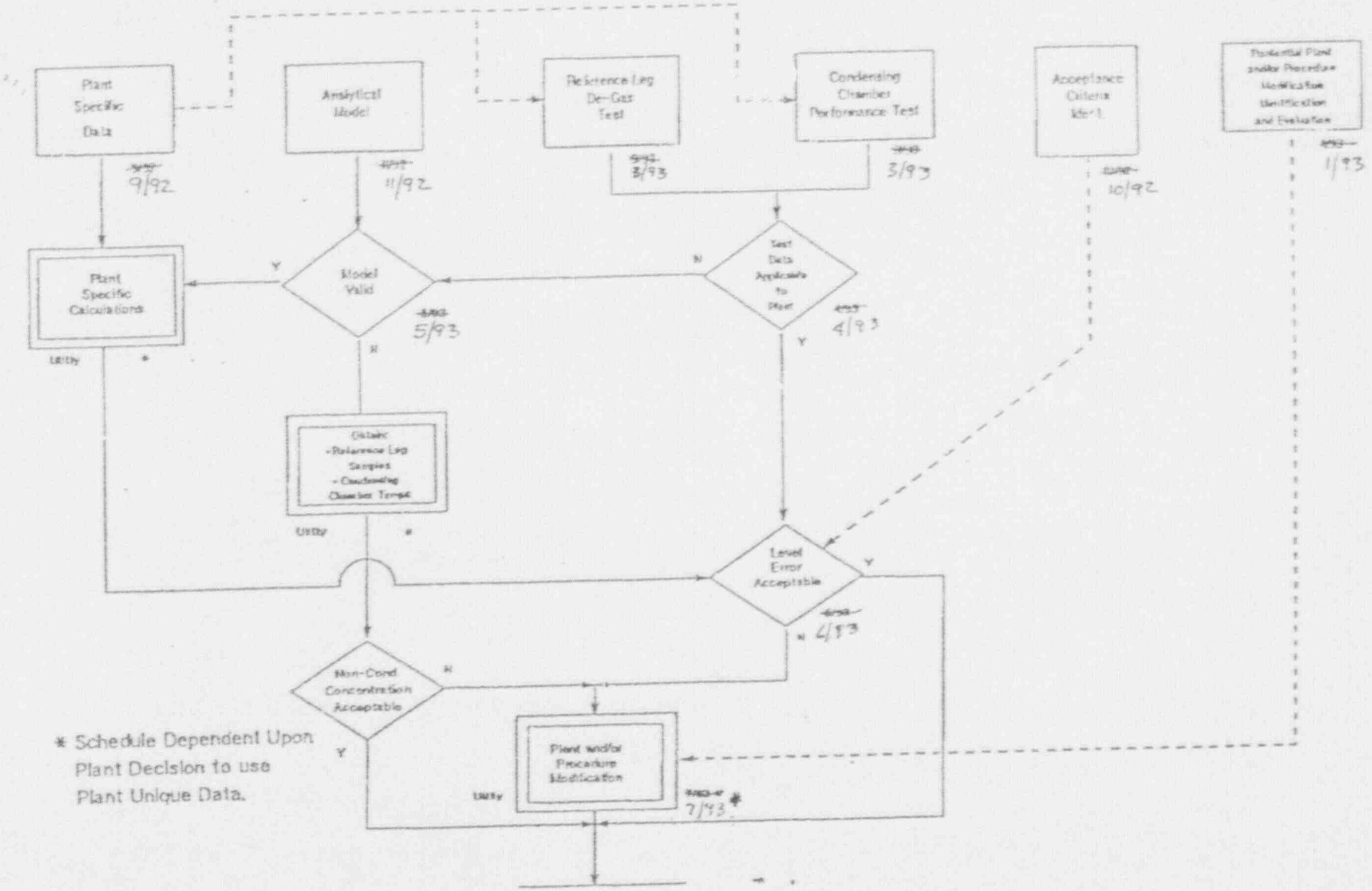
Evaluation of the plant-specific data will determine the need for plant or procedure modification.

Potential Modification Review

A review of potential changes to plant configurations or procedures to eliminate or accommodate errors will be evaluated. The type or extent of changes which may be necessary will depend upon the test and analytical results. This process will include work already completed or in progress at some plants.

Figure 1

BWROG Reactor Water Level Instrumentation Long Term Action Plan



* Schedule Dependent Upon Plant Decision to use Plant Unique Data.

Figure 2

Reference Leg De-Gas Test

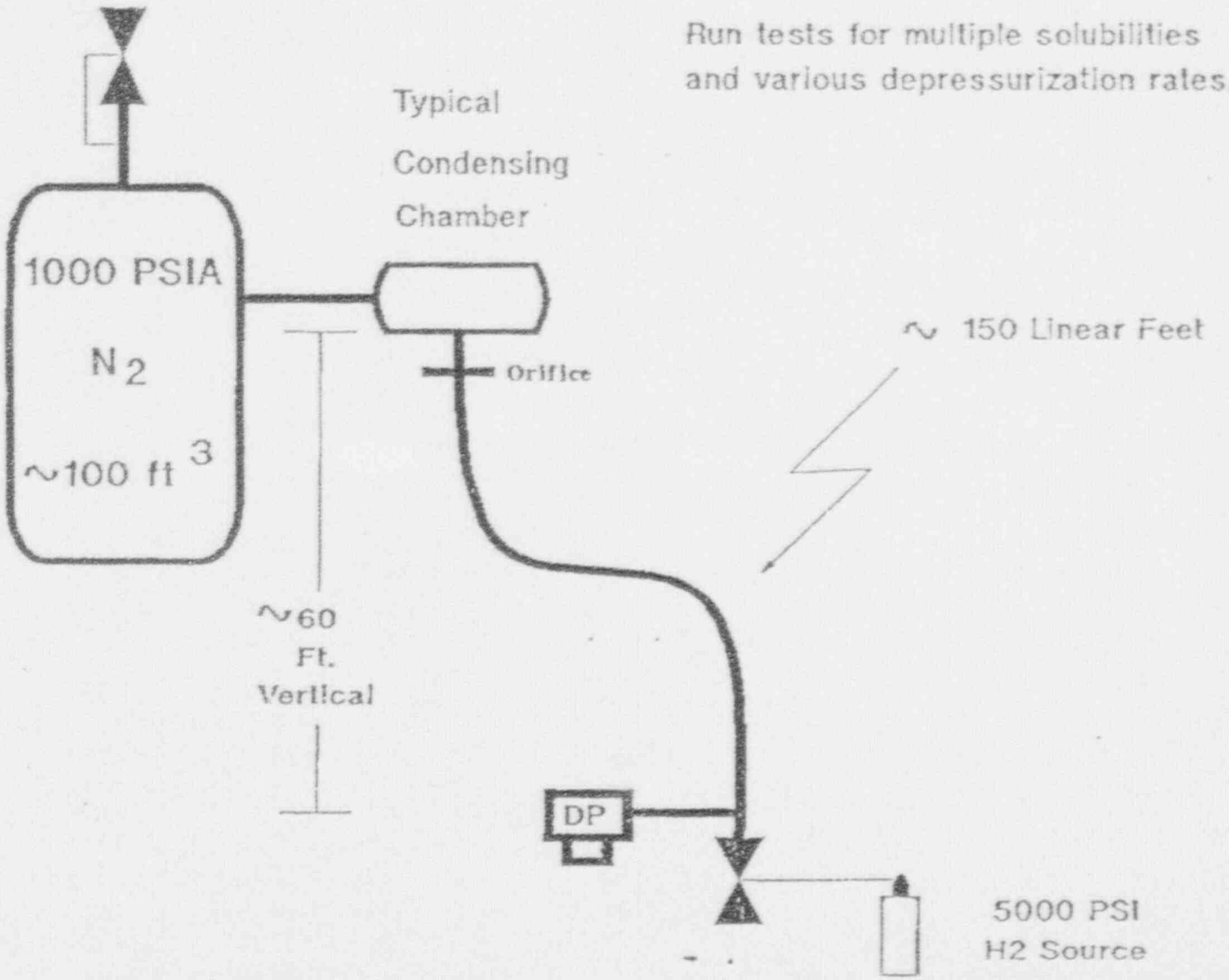


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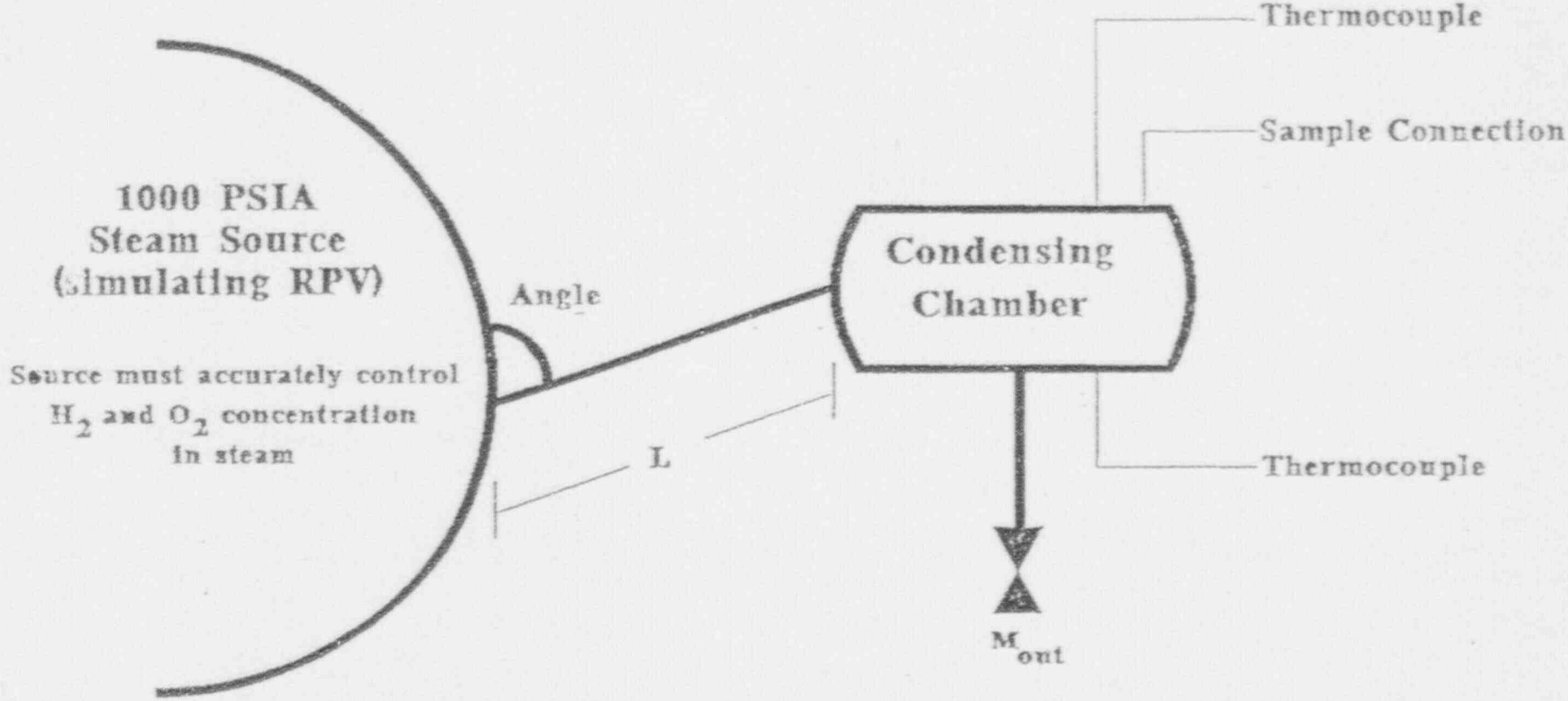
Figure 3

Condensing Chamber Performance Test



Test for inlet length and angle as well as CC geometry to establish data for the concentration of non-condensible gases entering the cold leg.

Enclosure 3

UTILITY COMMITMENT LIST

Utilities listed below communicated by noon on August 12 a commitment to make their response within 30 days of their receipt of the generic report.

Utility

Boston Edison Company
Carolina Power & Light Company
Cleveland Electric Illuminating Company
Commonwealth Edison Company
Detroit Edison Company
Entergy Operations
General Public Utilities Nuclear
Georgia Power Company
Gulf States Utilities Company
Illinois Power Comp: /
Nebraska Public Pov ır District
Niagara Mohawk Power Corporation
Northeast Utilities
Northern States Power Company
Pennsylvania Power & Light Company
Philadelphia Electric Company
Public Service Electric & Gas Company
Tennessee Valley Authority
Washington Public Power Supply System

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The Nuclear Industry has recently been investigating concerns centered around the performance/operability of reactor level instrument cold reference leg condensing chambers. The condensing chamber is a device which serves to maintain the reference leg of differential pressure type level measuring instrumentation at a known constant height based on its physical installation. (Please refer to attached simplified diagram). A constant level is maintained in the chamber via a continuous supply of nuclear steam condensate due to the chamber being a heat sink for steam which flows to it through the steam leg. The maintenance of a constant level reference leg is essential to the calibration basis and reliability of level measuring instrumentation. Unfortunately, non-condensible gases which compose part of the steam mixture are also free to enter the chamber. These noncondensible gases can exist as free gas in the chamber space (phenomenon 1) or as gas dissolved in the solution (phenomenon 2) which comprises the liquid reference leg.

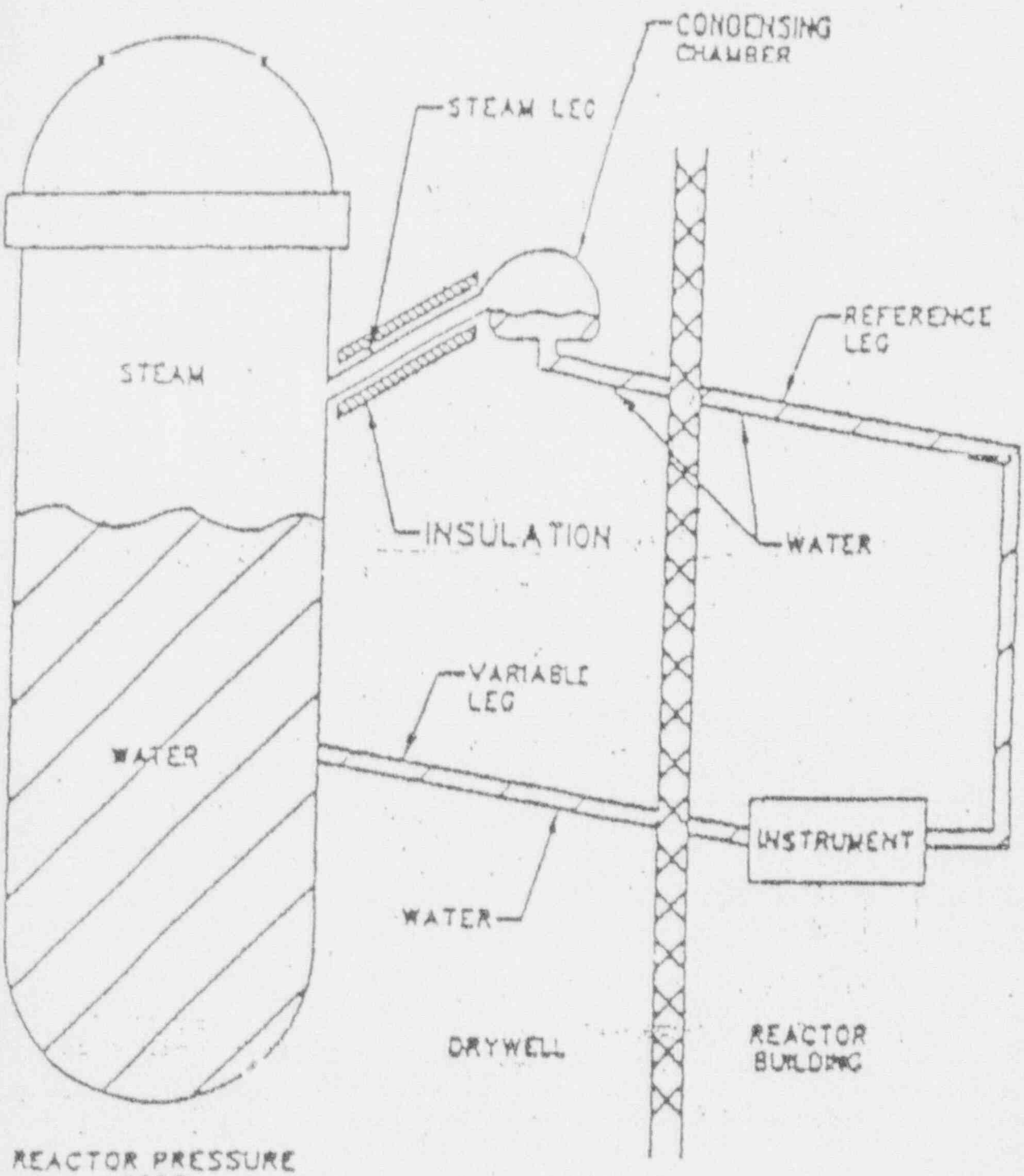
When free gas (phenomenon 1) is present it can impede, or in worst cases, prevent the ability of the chamber to condense steam by blanketing available surface area (for condensing) subsequently reducing the amount of condensate available for chamber level maintenance. Peach Bottom Unit 2 has actually experienced phenomenon 1 but in an aggravated sense. In two separate incidents over the past two years, PBAPS has shutdown due to inoperable levels channels identified during routine surveillance channel checks. It was determined that both events were due to a build-up of noncondensible gases coupled with reference leg leakage that became greater than the reduced condensing rate. The resultant degraded (lower) reference leg caused in both cases, the '2B' level instrument to read higher than actual level.

Phenomenon 2 involving saturation of the reference leg fluid with dissolved noncondensible gases is the issue of greatest debate and uncertainty at this time. These gases while in solution do not pose a problem for the reference leg, however if a rapid depressurization of this saturated fluid occurs, the subsequent degassing could expel part of the liquid in the leg. This would yield unstable false high readings from the level instruments. An illustrative example of this degassing phenomena can be seen when a shaken bottle of soda is opened. As in the soda bottle, the rate of reference leg depressurization is felt to greatly affect the amount of level that would be displaced by the degassing. It has been determined that the phenomena (1) experienced at PBAPS will not increase the probability of level errors during rapid depressurization. There is no conclusive real data available relative to this phenomena (2), but it is suspected that anomalies involving level indication errors would not occur until pressure goes below approximately 450 psig. The BWR Owners Group (BWROG) has reviewed existing design basis accident scenarios which lead to a lowering of reactor vessel water level and has concluded that automatic safety systems will be actuated at pressures well above 450 psig, even for postulated worst-case noncondensible gas concentrations in the reference legs. Therefore, there is confidence that all ECCS will initiate as they were designed to do. In addition, there are diverse signals, e.g. drywell pressure, that would also initiate ECCS for reactor water level lowering events. In conclusion, reactor water level instrumentation is considered operable.

At this time, the BWR owners group is devising a plan to conduct studies which will help bound the affects of this postulated phenomena on reactor water level indication error. Additionally, until this information can be properly evaluated and tested, there will be no revisions to procedures. However it should be noted that this condition may be relevant and should be considered during low pressure transients. As a result of the uncertainty associated with this issue, a heightened sensitivity to level indication abnormalities should be afforded during depressurizing below 450 psig.

SIMPLIFIED DIAGRAM OF A TYPICAL

WATER LEVEL MONITORING INSTRUMENT



THE NUMBER OF THESE INSTRUMENTS VARIES FROM PLANT TO PLANT. THE MINIMUM NUMBER IS TWO.