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

Hope Creek
50-354

OCT 2 1 1986

Barbara A. Curran, President
Board of Public Utilities
State of New Jersey
1100 Raymond Boulevard
Newark, New Jersey 07102

Dear President Curran:

This is in response to your letter, dated August 22, 1986, to Chairman Zech, regarding the discovery of incorrectly installed pressure sensing lines at the Hope Creek Nuclear Generating Station. These sensing lines are associated with the vacuum-breaker isolation valves which are designed to close on pressure in the torus and open when the torus is under vacuum, allowing the vacuum-breakers to equalize the pressure between the torus and the reactor building. The reversed sensing lines found at Hope Creek caused the isolation valves to operate opposite to this design.

In keeping with the Nuclear Regulatory Commission's mission to assure nuclear power plants are operated with proper regard for public health and safety, we not only inspect plant construction and operation, but provide careful followup on plant events of this type. Through detailed review of such events, we can determine whether the event has more far-reaching or generic significance. This aspect of our event review addressed the issue raised in your letter regarding the significance of this construction error in relation to the rest of the plant systems.

Following licensee discovery of the incorrectly installed pressure sensing lines on August 8, 1986, three inspectors from the NRC Region I (Philadelphia) office, including the assigned Senior Resident Inspector, conducted a detailed review of plant construction and testing to ascertain how the condition was created and whether the preoperational or startup test program should have detected the condition. The inspection report detailing this activity is attached for your information. As is the case with all Hope Creek inspection reports, a copy was also sent to the State of New Jersey, Department of Environmental Protection, Bureau of Radiation Protection.

The basic cause of the sensing line error was the preparation of an erroneous construction drawing on site. Since the same drawing was used as the basis for subsequent construction verifications, the error was not detected. In addition, testing techniques for the pressure instrument did not detect the piping error. Ultimately, a detailed review of an operating condition noted by the plant operators resulted in the discovery. The condition in question was the existence of a slight vacuum in the torus, which should have been automatically relieved by the vacuum-breakers.

NRC Region I will schedule a meeting with Public Service Electric and Gas Company to discuss the construction error in detail, including causes and corrective actions and any new information which may have been developed. Following the meeting, NRC will take appropriate enforcement action.

With regard to other undiscovered problems at Hope Creek, the licensee has taken actions to check other systems to ensure that no similar problems exist. Following the discovery of the reversed sensing lines, the licensee took a number of immediate corrective actions prior to restart of the Hope Creek reactor. These actions included: correctly aligning the sensor piping; reviewing and inspecting all similar pressure transmitter installations to ensure no other error or misapplication existed; conducting a check of all instrument valve positions to ensure correct alignment; reviewing all temporary plant modifications to ensure that no modification had impacted required system operability; and, instituting a program for identification and verification of instrument valve positions. During a meeting with NRC Region I management on August 22, 1986 to discuss their analysis and corrective actions, the licensee stated that no other similar piping errors had been identified. Our inspections have confirmed the adequacy of the corrective actions. In addition, the Hope Creek test program, including the current start-up testing, is intended to confirm that systems operate as designed. The NRC monitors the conduct of the overall test program to verify the adequacy of plant safety systems.

In addition to the above, more recent developments at the Hope Creek site in response to the loss of offsite power test results prompted the NRC to conduct a special inspection initiated by Region I. The charter of this inspection was to better understand and to assess the safety significance of the unanticipated test results. This initiative by the NRC, although not directly related to your August 22, 1986 letter, is an example of the measures taken by the NRC to ensure that established safety standards are met.

While I hope that the above information is responsive to your question, I would remind you that the Nuclear Regulatory Commission's responsibilities focus on the safety of the plant. Reliability and continuity of operation enter our area of interest peripherally, as they relate to reducing challenges to plant safety.

Should you have any further questions, please feel free to contact us again.

Sincerely,

Original signed by
Victor Stello

Victor Stello, Jr.
Executive Director for Operations

Attachment:
As stated
(Ltr fm W. Kane to Public Service
Electric & Gas Co. dated 10/6/86
enclosing Inspection Report No.
50-354/86-41)

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

06 OCT 1986

Docket Nos. 50-354

Public Service Electric and Gas Company
ATTN: Mr. C. A. McNeill, Jr.
Vice President - Nuclear
P. O. Box 236
Hancocks Bridge, NJ 08038

Gentlemen:

Subject: Inspection No. 50-354/86-41

This transmits the findings of a special safety inspection conducted by the NRC resident and region based inspectors on August 13 - September 2, 1986, at the Hope Creek Nuclear Power Station, Hancocks Bridge, New Jersey. The inspection consisted of a review of the causes for inoperability of the reactor building to suppression chamber pressure relief system identified on August 8, 1986. These findings were based on observation of activities, interviews and document reviews, and were discussed with Mr. R. Salvesen of your staff.

Areas examined during this inspection are described in the NRC Region I Inspection Report which is enclosed with this letter.

Based on the results of our inspection, we are concerned about the underlying reasons that let to operation of the Hope Creek plant with the reactor building to suppression chamber pressure relief system in an inoperable condition. The details of these concerns are described in the enclosed special inspection report. An enforcement conference will be scheduled in the near future, to be held in the Region I office to further discuss these concerns. At this conference you should be prepared to discuss your review of the concerns identified in the inspection report and corrective actions taken or planned.

The need for and nature of appropriate enforcement action relative to the issues identified in the enclosed report will be considered after this conference and will be the subject of separate correspondence at a later time. No response to this letter is required.

Your cooperation with us is appreciated.

Sincerely,

William F. Kane, Director
Division of Reactor Projects

Enclosure: NRC Inspection Report No. 50-354/86-41

8610160290

cc w/encl:
P. R. H. Landrieu, Vice President - Engineering (Newark)
R. J. Salvesen, General Manager, Hope Creek Operations
A. E. Giardino, Manager, Station Quality Assurance
W. H. Hirst, Manager, Joint Generation Projects Department, Atlantic Electric
L. A. Reiter, General Manager - Licensing and Reliability
Rebecca A. Green, Bureau of Radiation Protection
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
NRC Resident Inspector
State of New Jersey

While I hope that the above information is responsive to your question, I would remind you that the Nuclear Regulatory Commission's responsibilities focus on the safety of the plant. Reliability and continuity of operation enter our area of interest peripherally, as they relate to reducing challenges to plant safety.

Should you have any further questions, please feel free to contact us again.

Victor Stello, Jr.
Executive Director for Operations

Attachments: As stated

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Document Name:
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Requestor's ID:
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Author's Name:
Leif Norrholm

Document Comments:
Letter to Barbara A. Curran

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

50354-860808

Report No. 50-354/86-41

Docket 50-354

License NPF-57

Licensee: Public Service Electric and Gas Company

Facility: Hope Creek Generating Station

Conducted: August 13, 1986 - September 2, 1986

Inspectors: R. W. Borchardt, Senior Resident Inspector
D. J. Florek, Lead Reactor Engineer
J. A. Prell, Reactor Engineer

Approved: L. Norrholm, Chief Reactor, Projects Section 2B — 9/24/86 date

Inspection Summary:
Inspection on August 13, 1986 - September 2, 1986 (Inspection Report Number 50-354/86-41)

Areas Inspected: Special onsite inspection by the NRC resident and region based inspectors of the causes for the inoperability of the reactor building to suppression chamber pressure relief system. This inspection involved 93 hours by the inspectors.

Results: The inoperability of the reactor building to suppression chamber pressure relief system was identified as an apparent violation of Technical Specifications. Although this condition was licensee identified, it is viewed as significant because of the duration that this discrepancy existed and because a system which ensures primary containment integrity after certain postulated accident conditions was inoperable.

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DETAILS

1. Overview

At 11:45 a.m. on August 8, 1986, a reactor shutdown was commenced and an unusual event declared after it was determined by the licensee's staff that the reactor building to suppression chamber pressure relief system was inoperable and the plant was operating in violation of Technical Specifications. Technical Specification (T.S.) 3.6.4.2 does not allow plant operations in operational conditions 1, 2 or 3 with both Reactor Building to Suppression Chamber vacuum breaker assemblies inoperable, and therefore the plant was shut down as required by T.S. 3.0.3 which states that when a Limiting Condition for Operation is not met, except as provided in the associated action requirements, within one hour action shall be initiated to place the unit in an Operational Condition in which the specification does not apply. The Unusual Event was terminated at 3:35 p.m. when the unit entered operational condition 3 (Hot Shutdown). The unit entered operational condition 4 (Cold Shutdown) at 7:54 p.m.. The subsequent investigation into this event determined that a design drawing error made in 1983 during plant construction caused this system to be inoperable. In the event that a vacuum was created in the suppression chamber, the butterfly isolation valves in series with each vacuum breaker would have remained shut, which would have prevented the vacuum breakers from fulfilling their design safety function. This condition had existed since 1983, and remained undetected until August 8, 1986.

A design engineer's apparent misunderstanding of the reactor building to suppression chamber pressure relief system resulted in the reversed installation of the pressure differential transmitters intended to sense a vacuum in the suppression chamber. The as-installed configuration would have resulted in the 24 inch butterfly valve opening on high torus pressure rather than on high torus vacuum. A review of the construction program design controls identified no other similar discrepancies. A review of post construction activities indicates that once the original design error was made, no subsequent test would have identified the design error because both preoperational testing as well as surveillance testing isolated the sensing lines while a controlled pressure was applied directly to the pressure differential transmitter.

Despite the conditions described above, the ability of the 24 inch butterfly isolation valves to be manually operated from the control room was not affected.

This report provides a description of the reactor building to suppression chamber pressure relief system, discusses how the problem was identified, examines how the design error was made, and evaluates if other programs should have identified the deficiency earlier. The licensee's corrective actions are also discussed.

2. System Description

The reactor building to suppression chamber pressure relief (or vacuum breaker) system is designed to limit the presssure differential between the reactor building and the suppression chamber to less than 3.0 psid. This safety feature is intended to protect the suppression chamber from negative pressure loading in the event containment spray is inadvertently initiated after a loss of coolant accident. The reactor building to suppression chamber pressure relief (RBSCPR) system consists of two 24-inch independent vacuum relief assemblies, each providing a relief path from the reactor building air space to the suppression chamber. Each vacuum relief assembly consists of an outboard mechanical check valve and an inboard, normally closed, air operated butterfly valve. The two 24-inch mechanical check valves are designed to be fully open at 0.25 psid.

A pressure differential transmitter senses the pressure difference between the reactor building and the suppression chamber. When this differential reaches a specified limit, a pressure differential switch provides an input signal to a solenoid valve, associated with the butterfly valve, which energizes and directs gas from the primary containment instrument gas system (PCIGS) to open the butterfly valve. The path through the check and butterfly valves allows the reactor building atmosphere to enter the suppression chamber and equalize the pressure difference. When the pressure differential is reduced to a specified level, the solenoid valve de-energizes and the butterfly valve returns to the closed position. The check valve prevents the suppression chamber atmosphere from venting into the reactor building should the butterfly valve fail to close. Valve position switch contacts are monitored on each butterfly valve and each check valve, providing fully open and fully closed valve position indication in the main control room.

2.1 RBSCPR System Testability

The RBSCPR system controls and instrumentation are capable of being tested from sensors through actuating devices during normal power operation. Each sensor (pressure differential transmitter) can be valved out of service and functionally tested using an appropriate test source. This test verifies proper circuit operation from the transmitter input through the actuating device but does not test sensing line operability.

The calibration of the alarm units (pressure differential switches) can be checked from the appropriate cabinet in the control equipment room without initiating operation of the actuating device. When an alarm unit is placed "in test", an output is provided to illuminate an indicating light in the main control room to advise the control room operator of the "in test" status. This indicating light is automatically extinguished when the alarm unit is placed back in operation.

Each check valve and butterfly valve in the RBSCPR system can be individually tested from the main control room by the operation of an associated pushbutton test switch. When the pushbutton test switch is depressed, the solenoid valve associated with the valve being tested is energized and directs gas from the PCIGS to open the butterfly or check valve. When the pushbutton test switch is released, the solenoid valve de-energizes and the valve under test returns to its normal operating status. Satisfactory operation is determined by observation of the expected valve position indicating light patterns during the test.

2.2 Design Bases

As stated earlier, the RBSCPR system is designed to limit the presssure differential between the reactor building and the suppression chamber to less than 3.0 psid. The negative pressure differentials (negative corresponding to an inward loading) across the drywell and suppression chamber walls, as analyzed in the FSAR, are caused by the following events:

- cooling cycles;
- inadvertent containment spray actuation during normal operation; and,
- steam condensation following reactor coolant system (RCS) pipe ruptures with inadvertent containment spray actuation.

Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by heating and ventilating equipment. Inadvertent spray actuation during normal operation results in a more significant pressure transient and becomes important in sizing the suppression chamber-to-reactor building vacuum breaker assemblies. Steam condensation following RCS pipe ruptures with inadvertent containment spray actuation within the drywell results in the most severe pressure transients. Following an RCS rupture, the drywell atmosphere is purged to the suppression chamber free space, leaving the drywell full of steam. Subsequent condensation of the steam in the drywell can be caused either by ECCS spillage from the rupture or by inadvertent containment spray actuation following a LOCA.

Pressure transients within the drywell and the suppression chamber free space, due to inadvertent containment spray actuation for post-LOCA steam condensation, are evaluated in the FSAR. The results of containment depressurization transients are provided in FSAR Table 6.2-29. Details of the limiting transient, including the analytical models, assumptions, and methods used, are provided in FSAR Appendix 6A.

In response to this event the licensee has performed an additional analysis which shows the maximum allowable external pressure on the drywell to be in excess of 4 psi. With both vacuum breaker assemblies inoperable and assuming the worst case scenario, the negative pressure loading on the primary containment was calculated to be approximately 3.4 psid. The licensee therefore concluded that primary containment integrity would not have been compromised.

3. Problem Identification

On July 4, 1986, the control room operators observed that the torus pressure instrument indicated a pressure of -0.6 psig. The operators then took action to restore the torus pressure to within the Technical Specification limits of -0.5 psig to 1.5 psig and incident report 86-119 was generated to document this occurrence. As part of the incident report followup, the licensee investigated why the reactor building to suppression chamber vacuum breakers had not operated to relieve this presssure differential as designed. The presssure differential transmitters which control the inboard butterfly valves were verified to be within calibration and the butterfly valves were shown to be free to operate. Recent testing had also shown that the vacuum breaker check valves were operable. It was not until August 8, while performing a system review, that the system engineer identified a problem with the butterfly valves. His review indicated that a problem existed in the valve control logic. Once it was established that the RBSCPR system was inoperable, the unit was shut down to operational condition 3. The inoperability of the RBSCPR system during operational conditions 1, 2 and 3 is an apparent violation of Technical Specification Limiting Condition for Operation 3.6.4.2 (86-41-01).

On August 8, 1986, the Nuclear Safety Review Group was tasked to perform an independent assessment of the causes for the RBSCPR system's inoperability and to recommend corrective actions. The Safety Review Group directed that a complete system walkdown be performed, and during this evolution, the improperly installed pressure differential transmitter (PDT) tubing was identified. The high side of the PDT's were found to be connected to the torus and the low side to the reactor building, which is opposite of the arrangement required for proper system operation. The result of this error is that if a vacuum were drawn in the torus, the butterfly isolation valves would remain shut thereby preventing the vacuum breakers from relieving the pressure differential. Additionally, if a positive pressure was generated in the torus, the butterfly valves would open causing the vacuum breakers to be subjected to primary containment pressure. The inspector has subsequently reviewed the local leak rate test data for the vacuum breakers and determined that the results were within specifications and that primary containment leakage rates would have been acceptable even if the vacuum breakers were subjected to primary containment pressure.

The system walkdown identified two other discrepancies that impacted system operability. The PDT for one butterfly valve was found to be valved out of service. This would have prevented the valve from operating even if the PDT were installed correctly. Also, a sensing line for the same PDT had tape over its end, which may have impacted the PDTs ability to accurately sense reactor building pressure.

Although the system engineer's conclusion that a logic error was the cause for the system inoperability was not correct, it did serve to focus attention on the lack of control placed on the staple jumper configuration for Bailey 745 Alarm modules. This is a separate issue from the subject of this report and will therefore be discussed in inspection report 50-354/86-40.

4. Design and Construction Process

The pressure differential transmitters (PDT 5029 and 5031), used to sense suppression chamber vacuum, were designed and installed in accordance with the architect-engineer's field engineering program. Under this program, the conceptual system design comes from Bechtel's area office in San Francisco (SFAO). SFAO also approves and provides the instrument vendor prints and a "Design, Installation and Test Specification" (DITS) package for each major system. The DITS package provides a description of how each sub-system is to function, as well as design, installation and testing criteria.

After walking down the proposed routing of the system, the design engineer, using the above criteria, develops a diagrammatic field installation drawing (FSK). The FSK is independently reviewed by another design engineer and approved by a lead design engineer. The drawing is then turned over to the Field Engineering group who is responsible for working with craft personnel to install the system according to the drawing. Field Engineering uses a "red line" system for marking and identifying any necessary changes to the routing. Once installed, the "red lined" drawings are sent back to the design group, where any changes are reviewed by the design engineer and incorporated into an appropriate revision of the drawing. This revised drawing is subsequently reviewed by an independent engineer, the lead design engineer, and the lead field engineer. For subsequent revisions to the drawing, the design engineers review only the changes made to the drawing and not the whole drawing itself. Therefore, an error made in the original design drawing will not necessarily be identified during subsequent drawing revision reviews. Also, the Bechtel QA/QC department reviewed the drawings to verify that all administrative criteria had been met and not technical accuracy of the design.

Revision 0 of the FSK for PDT 5029 and 5031 were both completed on April 11, 1983. However, SFAO did not approve and issue the vendor prints for these instruments until July 1983. The vendor prints clearly show that the high pressure port of the PDT is on the left side and the low pressure port on the right side of the installed sensor. Revision 0 of the FSK for PDT 5029 correctly showed the routing of the tubing to the PDT. However, Revision 0 of the FSK for PDT 5031, incorrectly routed the tubing for the high and low pressure ports. The same designer and reviewer were responsible for both FSKs.

The tubing installation and rework for PDT 5029 was completed in October 1984. At this time, the tubing lines were reversed per the field engineer's instructions. This reversed installation was then approved and incorporated into Revision 1 of the FSK. A third design engineer reviewed and approved Revision 1, but the same engineer who initially checked the earlier revision also checked this revision.

The tubing installation for PDT 5031 was not completed until April 1985. This installation was made per the drawing, which was subsequently determined to be incorrect. Revision 1 and all subsequent revisions of this drawing show the tubing being routed to the wrong ports of the PDT.

Each of the design engineers involved with this event had significant experience in design work at nuclear power plants, and, it appears that, after reviewing the Bechtel design engineering and installation process, that sufficient engineering design controls were established. In addition, no other similar discrepancies were identified.

5. Review of Preoperational Test Results Data

The inspector reviewed portions of preoperational tests PSSUG-PTP-GS-1, Containment Atmosphere Control System, and PSSUG-PTP-GP-1, Primary Containment Integrated Leak Rate Test, to determine if preoperational testing contained precursors of the vacuum breaker problem or was inappropriately performed. Specifically, the inspector reviewed the methodology of performing the preoperational testing on 1-GS-PDT-5029 and 1-GS-PDT-5031 and reviewed the containment integrated leak rate test (CILRT) preoperational test to determine if the 24 inch butterfly valve opened during the test. The inspector concluded that the methodology utilized to perform the functional testing of the pressure differential transmitter through operation and cycling of the 24 inch butterfly valve was not capable of identifying that the sensing lines to the PDTs were installed in the reverse manner. In addition, during the CILRT, the PDTs were isolated per procedure so that no data or conclusion could be drawn from this test relative to the sensing line configuration.

The methodology used to test the PDTs in the preoperational test program required connecting a variable pressure test source to the PDT and increasing the pressure until the valve opened. Interviews with various plant personnel indicated that test personnel are instructed to always utilize the high side of the PDT to impose pressure for testing. This would require isolating the reactor building and torus sensing lines to the PDT, and installing a pressure source to the high side of the PDT. The required pressure differential is then applied via a high side test connection. Since the butterfly valve control logic assumes the high side of the transmitter is connected to the reactor building air space, as the test pressure is increased the butterfly valves should open. (NOTE: a positive pressure to the high side (reactor building) is equivalent to a vacuum on the low side (torus)). As installed, the tubing on the high side of the PDT was incorrectly connected to the torus.

The preoperational test (PSSUG-PTP-GS-1) was performed several times. Several test exceptions were identified due to recalibration of the pressure differential transmitter. No concerns were identified regarding the configuration of the sensing lines.

The inspector also reviewed several Test Pack Release Documents for PDT-5029 (TPR-GSC - 9, 12, 22, 156, 186, 323, 441 and 444) and for PDT-5031 (TPR-GSC - 30, 72, 104, 187, 190, 308, 414, 420). The method used to perform these calibrations was similar to the actual preoperational test. The sensing lines were isolated and a regulated pressure supply connected to the pressure differential transmitter. The inspector determined that none of the testing identified the problem.

6. Review of PDT Setpoint Calculations

The inspector reviewed the following calculations to determine if the calculations used the "as-built" configuration of the sensing lines:

Calculation 113	Reactor Building to Suppression Chamber Differential Pressure High, dated January 10, 1985
Calculation GS-18	Process Setpoints for Suppression Pool to Reactor Building Breakers, dated August 5, 1986
Calculation SC-GS-0101	Setpoint Calculation for Reactor Building Atmosphere Control, dated June 7, 1986

Setpoint register J040Z, Revision 1 dated January 15, 1986, was also reviewed. No indication existed that the calculations of setpoints assumed anything other than calculating pressure differential with high pressure on the reactor building side relative to the torus.

7. Surveillance Procedure Review

The following surveillance procedures were reviewed to determine if they identified the sensing line installation error:

- IC-SC.GS-008 Containment Atmosphere Control - Channel A PDT-5029 (Revision 0 dated September 4, 1986, and Revision 1 dated May 27, 1986)
- IC-CC.GS-006 Containment Atmosphere Control Division 1 Channel PD-5029 (Revision 0 dated October 4, 1985, Revision 1 dated June 2, 1986 and Revision 2 dated June 11, 1986)
- IC-CC.GS-005 Containment Atmosphere Control Channel B P-5031 (Revision 0 dated October 4, 1985, Revision 1 dated June 2, 1986 and Revision 2 dated June 2, 1986)

The methodology to perform the surveillances is similar to that of the preoperational test, in that the sensing lines are isolated for the PDT calibration and a controlled pressure device is used to calibrate the PDT. Similarly for the logic portion of the circuit, the PDT is electrically isolated and a digital calibrator is utilized. The inspector concluded that the surveillance procedures would not determine that the sensing lines to the PDT were reversed.

8. Licensee's Corrective Actions

As a result of the discrepancies identified during the review of this event, the licensee initiated a number of corrective actions. These actions can be grouped into three categories, listed by the time frame of their completion.

1. Immediate (Prior to Unit Startup)

   - A Design Change Package was completed, correcting the PDT tubing error for PDT 5029 and 5031.

   - A review of Q-Listed vacuum applications for differential pressure transmitters was conducted to search for other tubing errors or misapplications. No problems were identified.

   - Instrument and Control personnel conducted a complete instrument valve lineup verification for all instruments in the reactor building. One discrepancy was identified, in that a MSIV sealing system pressure transmitter used for an alarm annunciator was found isolated. This was corrected.

   - All Q, F and R Temporary Modifications were reviewed to verify that safety evaluations and unreviewed safety question determinations had been made. This action was taken to provide added assurance that no changes had been made to the plant design that could possibly impact system operability. The licensee identified no additional problems.

   - SORC Meeting 86-196, held on August 11, 1986, approved a program for identification and position verification of Q-Listed Instrument Valves.

2. Short Term Recommendations

   - Verify that all non Q-Listed vacuum-pressure differential pressure applications have correct instrument tubing routing installation to perform their intended design function.

   - Develop a schedule for the expeditious conversion of existing temporary modifications to Design Change Packages.

- Revise Administrative Procedure AP-13 to require performance of a safety evaluation for all temporary modifications.

3. Long Term Recommendations

- Terminate the practice of using temporary modifications in lieu of DCPs for permanent plant modifications.

- Establish a policy of completing the incident report, root cause assessment/draft LER within 20 days of the incident.

- Develop, implement, and maintain a formal program to identify and document the design basis for Hope Creek.

On August 22, 1986, the Region I staff met with the licensee at the NRC Region I office in King of Prussia, Pennsylvania to discuss this event and the licensee's corrective actions. Enclosure 1 to this report identifies the attendees and the information provided by PSE&G.

9. Exit Interview

The inspectors met with licensee and contractor personnel periodically and at the end of the inspection report to summarize the scope and findings of their inspection activities. Written material was not provided to the applicant during the exit.

Based on Region I review and discussions with the licensee, it was determined that this report does not contain information subject to 10 CFR 2 restrictions.

The inspector summarized that the as-built configuration of the reactor building to suppression chamber pressure relief system resulted in the system not being operable and was an apparent violation of Technical Specification 3.6.4.2. Enforcement pertaining to this issue would be addressed separately from the inspection report.

Enclosure 1

August 22, 1986 Meeting Between PSE&G and NRC Region I

List of Attendees

Name	Title	Organization
T. Murley	Regional Administrator	NRC Region I
W. Kane	Director, Division of Reactor Projects	NRC Region I
R. Summers	Project Engineer	NRC Region I
S. Collins	Deputy Director, Division of Reactor Projects	NRC Region I
D. Allsopp	Resident Inspector, Hope Creek	NRC Region I
R. Gallo	Chief, Reactor Projects Branch 2	NRC Region I
T. Kenny	Senior Resident Inspector, Salem	NRC Region I
S. Ebneter	Director, Division of Reactor Safety	NRC Region I
J. Durr	Chief, Engineering Branch	NRC Region I
D. Florek	Lead Reactor Engineer	NRC Region I
P. Eapen	Chief, Quality Assurance Section	NRC Region I
C. A. McNeill	Vice President - Nuclear	PSE&G
R. Salvensen	General Manager - Hope Creek Operations	PSE&G
B. A. Preston	Manager - Licensing & Regulation	PSE&G
G. Peet	Lead I&C System Engineer	PSE&G/System
W. Braver	Principal Safety Review Engineer	PSE&G
R. Burricelli	General Manager Engineering and Plant Betterment	PSE&G
J. Mackinnon	General Manager Nuclear Safety Review	PSE&G
D. Sullivan	Resident Project Engineer	Bechtel

TECHNICAL SPECIFICATION 3.6.1.6

"DRYWELL AND SUPPRESSION CHAMBER INTERAL PRESSURE SHALL BE MAINTAINED BETWEEN -0.5 AND +1.5 PSIG".

BASES

"THE LIMITATIONS ON DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE ENSURE THAT THE .....EXTERNAL PRESSURE DIFFERENTIAL DOES NOT EXCEED THE DESIGN MAXIMUM EXTERNAL PRESSURE DIFFERENTIAL OF 3 PSID."

EVENTS

- 7/4/86 - TORUS PRESSURE -0.6 PSIG
- 7/31/86 - TORUS PRESSURE -0.5 PSIG
- 8/8/86 - FORCED SHUTDOWN
  - EXCEEDED LCO 3.6.4.2
  - BOTH REACTOR BUILDING - SUPPRESSION CHAMBER BUTTERFLY ISOLATION VALVES DECLARED INOPERABLE

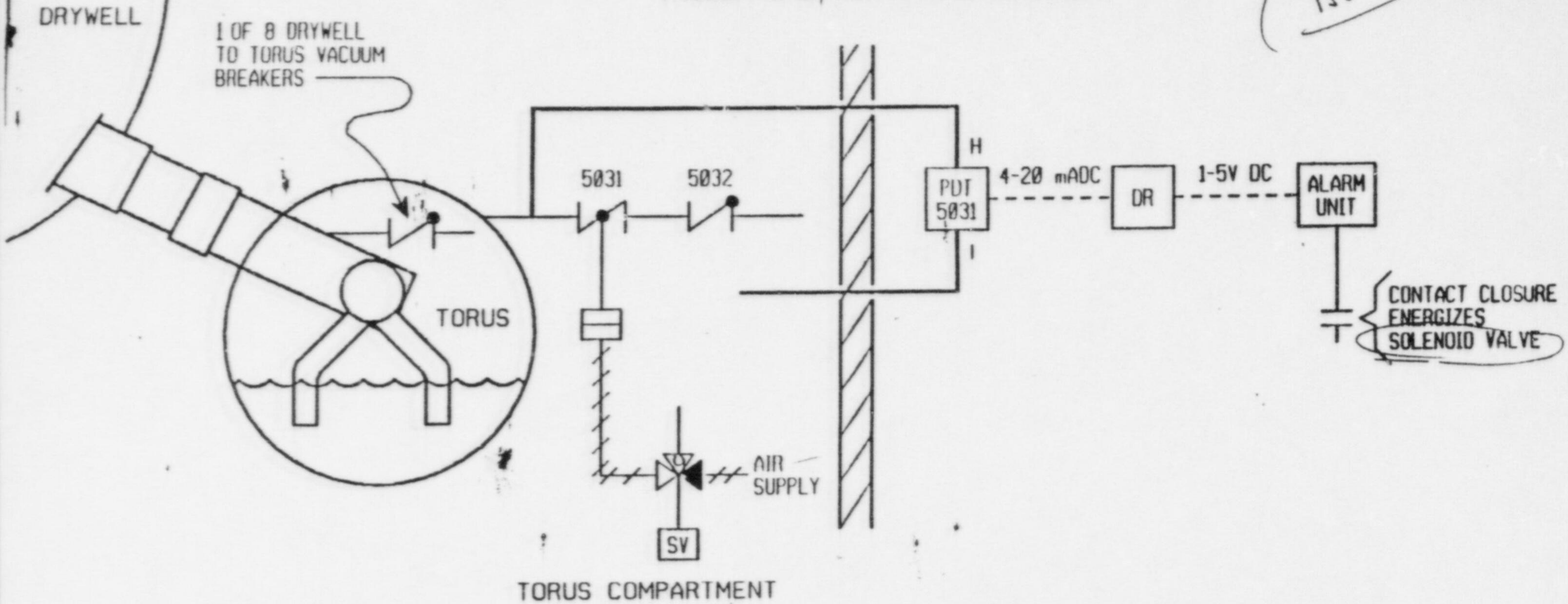
INVESTIGATION

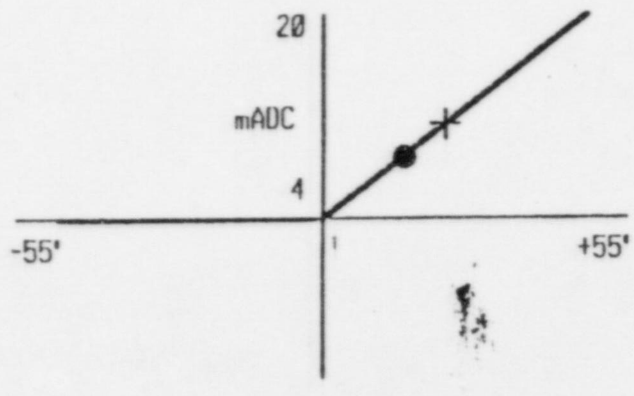
- 8/8/86 - TASK FORCE COMMISSIONED BY VICE PRESIDENT TO INVESTIGATE FORCED SHUTDOWN

- 8/13/86 - TASK FORCE REPORT CONCLUDES ROOT CAUSE TO BE DESIGN DEFICIENCY (ISOMETRIC DRAWINGS SHOW TUBING TO PDT REVERSED).

TORUS TO REACTOR BUILDING VACUUM BREAKER SYSTEMS/LINES

As found





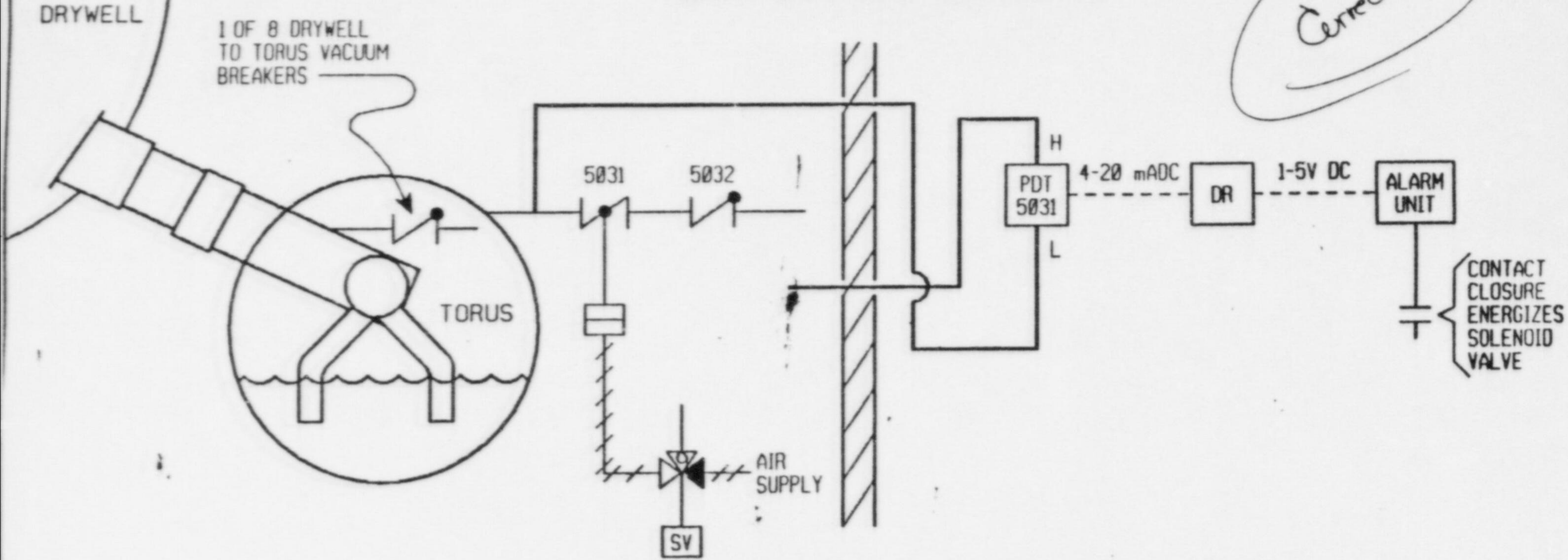
RESPONSE CURVE FOR INSTRUMENT LOOP

× = TRIP = + 4.99' (+ .18 PSID)

● = RESET = + 4.71' (+ .17 PSID)

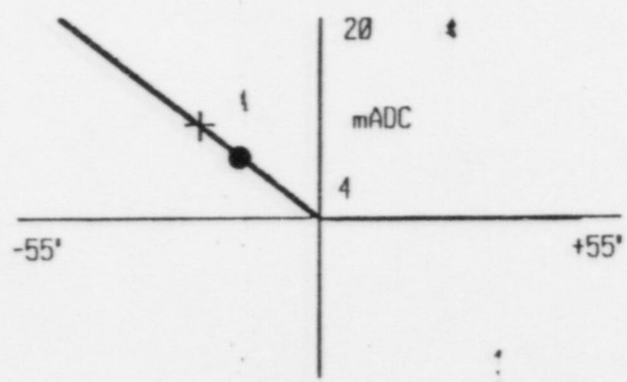
TORUS TO REACTOR BUILDING VACUUM BREAKER SYSTEMS/LINES

Corrected



TORUS COMPARTMENT

CORRECTIVE ACTION



× = TRIP = - 4.99' (- .18 PSID)

● = RESET = - 4.71' (- .17 PSID)

RESPONSE CURVE FOR INSTRUMENT LOOP WITH CORRECTIVE ACTION

OBSERVATIONS MADE AS A RESULT OF TASK FORCE INVESTIGATION

- SYSTEM TESTING PROGRAM WOULD NOT HAVE UNCOVERED REVERSAL OF THE SENSING LINES. TESTING WAS FULL INSTRUMENTATION CHANNEL TESTING VERSUS FULL SYSTEM OPERATIONAL TEST SINCE FULL SYSTEM TESTING WAS IMPRACTICAL.

- DECREASING TREND OF SUPPRESSION CHAMBER PRESSURE OUTSIDE OF NORMAL LIMITS AND FAILURE OF VACUUM BREAKERS TO CORRECT PROBLEM ON 7/4/86 AND 7/31/86 WAS NOT IDENTIFIED BY OPERATORS.

- COMPREHENSIVE REVIEW PROCESS INHERENT IN THE HOPE CREEK INCIDENT REPORT PROGRAM LED TO DISCOVERY OF THE DEFICIENCY. HOWEVER, REVIEW OF THE 7/4 EVENT WAS NOT YET COMPLETE, WHICH PRECLUDED PLANT MANAGEMENT FROM BEING MADE AWARE OF THE SIGNIFICANCE OF THE INCIDENT REPORT IN A TIMELY MANNER.

OBSERVATIONS MADE AS A RESULT OF TASK FORCE INVESTIGATION (cont.)

- T-MOD APPROVED FOR INSTALLATION TO CORRECT THE DESIGN DEFICIENCY WOULD NOT, IN FACT, HAVE CORRECTED THE DEFICIENCY.
- PDT-5031 FOUND TO BE ISOLATED AT MANIFOLD DURING 8/8 AS-BUILT VERIFICATION.
- THE ROLE OF DITS RELATIVE TO THE DESIGN AND LICENSING REQUIREMENTS IS NOT APPARENT; THEREFORE, IT IS NOT CLEAR WHAT DOCUMENTS SHOULD BE USED TO DETERMINE THE DESIGN BASIS FOR THIS SYSTEM.

RECOMMENDED CORRECTIVE ACTION THAT HAD TO BE TAKEN PRIOR TO RESTART

1. COMPLETE INSTALLATION OF THE DCP TO CORRECT THE INSTRUMENT TUBING ERROR AT GS-PDT-5029 AND 5031.

2. DETERMINE THAT OTHER Q-LISTED VACUUM-PRESSURE DIFFERENTIAL PRESSURE APPLICATIONS HAVE CORRECT INSTRUMENT TUBING ROUTING INSTALLATIONS TO PERFORM THEIR INTENDED DESIGN FUNCTIONS.

3. VERIFY THAT ALL EXISTING Q, F, OR R TEMPORARY MODIFICATIONS HAVE A SAFETY EVALUATION AND UNREVIEWED SAFETY QUESTION DETERMINATION.

4. OBTAIN SORC APPROVAL OF A PROGRAM AND SCHEDULE FOR IDENTIFICATION AND POSITION VERIFICATION OF Q-LISTED INSTRUMENT VALVES.

SHORT TERM RECOMMENDATIONS

1. VERIFY THAT ALL NON-Q LISTED VACUUM-PRESSURE DIFFERENTIAL PRESSURE APPLICATIONS HAVE CORRECT INSTRUMENT TUBING ROUTING INSTALLATION TO PERFORM THEIR INTENDED DESIGN FUNCTION.

2. DEVELOP A SCHEDULE FOR THE EXPEDITIOUS CONVERSION OF EXISTING TEMPORARY MODIFICATIONS TO DCP'S.

3. REVISE ADMINISTRATIVE PROCEDURE AP-13 TO REQUIRE PERFORMANCE OF A SAFETY EVALUATION FOR ALL TEMPORARY MODIFICATIONS.

LONG TERM RECOMMENDATIONS

1. TERMINATE THE PRACTICE OF USING TEMPORARY MODIFICATIONS IN LIEU OF DCPS FOR PERMANENT PLANT MODIFICATIONS.

2. ESTABLISH A POLICY OF COMPLETING THE INCIDENT REPORT, ROOT CAUSE ASSESSMENT/ DRAFT LER WITHIN 20 DAYS OF THE INCIDENT.

3. DEVELOP, IMPLEMENT, AND MAINTAIN A FORMAL PROGRAM TO IDENTIFY AND DOCUMENT THE DESIGN BASIS FOR HOPE CREEK.

GENERAL RECOMMENDATION

ASSURE DESIGN REVIEWS ARE CONDUCTED IN SUFFICIENT DEPTH AND THE DESIGN BASIS FOR THE SYSTEM BEING MODIFIED IS CLEARLY UNDERSTOOD PRIOR TO MAKING CHANGES:

<u>Enclosure 2</u>

<u>References/Drawings</u>

During the course of this inspection the following references were utilized:

P&ID M-57-1, Revision 14 Containment Atmosphere Control

FSK-JD-1303-1-001-1, Revision 8

FSK-JD-1303-1-001-2, Revision 5

FSK-JD-1803-1-010-1, Revision 6

FSK-JD-1803-1-010-2, Revision 5

FSK-JD-1803-1-010-1, Revision 0

FSK-JD-1803-1-010-1, Revision 1

FSK-JD-1803-1-010-2, Revision 0

FSK-JD-1803-1-010-2, Revision 1

FSK-JD-1303-1-001-2, Revision 0

FSK-JD-1303-1-001-2, Revision 1

PSSUG-PTP-GS-1, Containment Atmosphere Control System Preoperational Test

PSSUG-PTP-GP-1, Primary Containment Integrated Leak Rate Test

Test Package Release Documents for PDT-5029, TPR-GSC 12, 22, 156, 186, 323, 441, 444

Test Package Release Documents for PDT-5031, TPR-GSC-30, 72, 164, 187, 190, 308, 414, 420

Containment Atmosphere Control Calculations:

GS-113 Reactor Building to Suppression Chamber Differential Pressure High, dated January 10, 1985

GS-18 Process Setpoints for Suppression Pool to Reactor Building Breakers, dated August 5, 1986

SC-GS-0101 Setpoint Calculation for Reactor Building Atmosphere Control, dated June 7, 1986

Setpoint Register J0402, Revision 1

IC-SC-GS-008, Revision 0 and Revision 1 Containment Atmosphere Control - PDT 5029

IC-CC-GS-006, Revision 0, Revision 1, Revision 2 Containment Atmosphere Control (C.A.C.) Division 1 PD 5029

IC-CC-GS-005, Revision 0, Revision 1, Revision 2 C.A.C. PDT-5031

10855-D3.40, Revision 4 - Design, Installation and Test Specification for Containment Atmosphere Control System for the Hope Creek Generating Station

QC File No. 3G6-M57-1-1, QCIR No. - FSK-JD-1303-1-001-1-2-I1.10

QC File No. 3G6-M57-1-1, QCIR No. - FSK-JD-1803-A-010-1-1-2

QC File No. 3G6-M57-1-1, QCIR No. - FSK-JD-1303-1-002-1-1C-I1

QC Instruction No. 10855/I-1.10, Revision 3, Installation of Instruments

Specification 10855-J-825(Q), Revision 8, Technical Specification for Instrument Installation

SWP/P-057, Revision 1, Specific Work Plan/Procedure for Installation of Instrumentation

SWP/P-010, Revision 19, Specific Work Plan/Procedure for Field Design Approval and Control

SWP/P-J-101, Revision 9, Specific Work Plan/Procedure for Instrument Field Design, Materials Installation, Surveillance/Inspection