U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-334/85-03

Docket No. 50-334 License No. DPR-66

Licensee: Duquesne Light Company
One Oxford Center
301 Grant Street
Pittsburgh, PA 15279

Facility Name: Beaver Valley Power Station, Unit 1

Inspection at: Shippingport, Pennsylvania

Inspection Conducted: January 8 - 15, 1985

Inspectors: Wm. Troskoski 1/18/85
W. M. Troskoski, Senior Resident Inspector date signed

Wm Troskoski /for 1/18/85
D. M. Johnson, Resident Inspector date signed

Approved by: L. E. Tripp 1/19/85
L. E. Tripp, Chief, Reactor Projects date signed
Section No. 3A, Reactor Projects
Branch 3

Inspection Summary: Inspection No. 50-334/85-03 on January 8 - 15, 1985.

Areas Inspected: Special inspection by the resident inspectors (22 hours) to review: (1) the circumstances surrounding the failure to establish containment integrity prior to entering hot standby (Mode 4) on December 23, 1984, during reactor startup following the fourth refueling outage; and (2) incorrect pressurizer high level reactor trip setpoints, in excess of Limiting Safety System Settings. The inspectors interviewed Operations personnel and reviewed station logs and startup checklists.

Results: Three potential violations were identified: failure to establish containment integrity during startup, failure to immediately notify the Commission of the reportable event pursuant to 10 CFR 50.72, and failure to set the pressurizer high level reactor trip setpoints within the Allowable Value of Technical Specification 2.2, Limiting Safety System Settings.

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DETAILS

1. Persons Contacted

During the course of this special inspection, the following principal licensee personnel were contacted or interviewed:

Licensee Personnel

T. D. Jones, General Manager, Nuclear Operations
W. S. Lacey, Plant Manager
L. G. Schad, Operations Supervisor
J. D. Sieber, General Manager, Nuclear Services

The inspector also contacted other licensee personnel during this inspection.

2. Introduction - Containment Integrity

In the course of plant startup activities at the conclusion of the fourth modification and refueling outage, hot standby (Mode 4) conditions were established on December 23, 1984. A containment vacuum of 9.9 psia had been previously achieved, but difficulties were encountered in maintaining pressure below the technical specification Maximum Allowable Pressure (MAO). Licensee investigation identified valve packing leaks on two safety injection valves (MOV-SI-860A, B) and a mispositioned one inch casing drain valve (RS-113) on the A outside recirculation spray pump (RS-P-2A), which should have been closed. After closing RS-113 and tightening the packing on MOV-SI-860 A and B, the plant was brought to cold shutdown (Mode 5) to avoid entering the MAO action statement. This allowed use of the containment steam jet air ejector to redraw containment vacuum. The open condition of valve RS-113 was not recognized as a violation of containment integrity (Technical Specification 3.6.1.1) and was not reported pursuant to 10 CFR 50.72, Immediate Notification Requirements.

Initial recognition of a potential containment integrity violation was established at the exit meeting for NRC Inspection Report 50-334/84-33, on January 7, 1985. This special inspection was conducted to review the circumstances surrounding this event.

2.1 System Design Characteristics

The containment at BVPS Unit 1 was designed and built by the Stone and Webster Engineering Company (A-E), and consists of a reinforced concrete structure that is maintained at a subatmospheric pressure during plant operation. Technical Specification 3.6.1.4 specifies the Maximum Allowable Operating (MAO) pressure permitted when the plant is in Modes 1 thru 4. The MAO is obtained through a correlation of river water and RWST temperature and has a range of 8.9 to about 10.5 psia. Initial containment vacuum is achieved through the use of a steam jet air ejector. Containment integrity requirements specified in TS 3.6.1.1 require this penetration to be manually isolated in Modes 1 thru 4. During normal operation, two mechanical vacuum pumps (5 SCFM, each) are provided to compensate for air inleakage of up to 400% of the designed value.

Two recirculation spray systems each consisting of a pump, heat exchanger, and associated piping, are used to return and maintain containment pressure subatmospheric and provide long term decay heat removal during a design basis accident. The inside recirculation system is located entirely inside containment. The two pumps associated with the outside recirculation system (RS-P-2A, B) are located in the Safeguards Building, which in effect extends the containment boundry to the pump casings and seals. The suction line associated with each pump originates at the containment sump and is normally dry.

The Supplementary Leak Collection and Release System collects any air leakage to the Safeguards Building and processes it through the main filter banks prior to release. The FSAR takes no credit for this system in calculating releases during postulated accidents.

2.2 Management Control Requirements

A. Documents Reviewed

-- BVPS Station Administrative Procedure (SAP)-4 Operations.

-- BVPS OM Chapter 1.48, Conduct of Operations.

-- BVPS OM Chapter 1.50, Station Startup

-- BVT 1.3-1.47.7 Containment Isolation Valve Leakage Test Connection Verification.

-- Shift Operating Reports, December 23 - 26, 1984.

-- Nuclear Control Operators Logs, December 23 - 26, 1984.

B. Requirement Bases

The administrative requirements for the control and operation of plant equipment important to safety originate in the BVPS Unit 1 Quality Assurance Program as described in Appendix A to the updated Final Safety Analysis Report. Station Administrative Procedures (SAP), Chapter 4, Operations, defines the methods and responsibilities for conducting plant operational activities in accordance with those QA commitments, and OM Chapter 1.48, Conduct of Operation, provides the specific implementing instructions.

A basic operational precept referenced in SAP-4 is that the Nuclear Shift Supervisor must be continuously aware of the status of all systems and equipment important to safety. The following are requirements to ensure this during plant startup.

1. OM Chapter 1.50, Station Startup, contains checklists and procedures that are required to be completed prior to entering an elevated operational mode. To startup from an extended refueling outage, system lineups are performed in accordance with the Station Startup Prerequisite Checklist.

2. SAP-4 requires the Nuclear Station Operating Supervisor (NSOS) to prepare the Startup Prerequisite Checklist for approval by the Station Superintendent. This list does not replace the OM Chapter 1.50 Startup Checklists, but supplements them to assure that such items as maintenance work requests, clearances, deficiency lists, procedure changes, etc., are closed out at the appropriate time.

3. To comply with NUREG-0737, Item IC6, Verification of Operating Activities, verbal instructions from the NSOS required a double verification of safety system lineups listed in OM Chapter 1.50.

4. OM Chapter 1.48 provides equipment clearance instructions. For safety-related equipment, the clearance form requires a double verification for returning the system or component to its normal alignment.

2.3 <u>Sequence of Events</u>

Resident Inspector discussions with shift personnel and review of shift logs, startup procedures and checklists determined that the following sequence of events led to a breakdown of the licensee's management control system used to assure the establishment of containment integrity prior to plant heatup.

-- On December 8, 1984, RS-P-2A was placed on clearance for replacement of the pump seals. The pump casing drain valve, RS-113, was opened as part of the clearance. The plant was in cold shutdown and making preparations for startup.

-- Pump seal replacement was completed on December 13, 1984. The maintenance mechanic signed the clearance "off log only" in the clearance log book. This allows the mechanic to energize or operate equipment which has been electrically or mechanically isolated by a clearance, for component testing.

-- Double lineups were performed per OM Chapter 1.13, Containment Depressurization System, on December 14, 1984. Both lineups identified RS-113 as being open and on clearance.

-- The equipment clearance log review checkoff was moved from a Mode 5 to a Mode 2 requirement on the Startup Prerequisite Checklist.

-- The containment steam jet air ejector was put on line at 1719 hours on December 22, 1984, and containment vacuum of 9.9 psia was achieved at 1230 hours on December 23, 1984. At this time, the licensee believed that all containment integrity requirements of TS 3.6.1.1 were met.

-- Hot Standby (Mode 4) conditions were established at 1852 hours, December 23, 1984.

-- Containment vacuum was tracked on special logs by the Shift Technical Advisors to ensure that the MAO pressure referenced in Technical Specification 3.6.1.4 would not be exceeded as primary plant temperature was increased.

-- Logs of containment pressure showed a gradual increase of about 0.1 psia per shift with both mechanical vacuum pumps in operation.

-- Subsequent investigation identified a packing leak on MOV-SI-860 A and B which was repaired. At 1000 hours on December 23, 1984, RS-113 was found open and immediately closed.

-- Due to the fact that containment pressure had increased to the point where the containment steam jet air ejector would be needed to re-establish containment vacuum, a cooldown to Mode 5 was initiated on December 26 and Mode 5 was entered on December 27 at 0634 hours.

-- A containment vacuum was re-established on December 27 at 1032 hours and Mode 4 was re-entered at 1426 hours.

2.4 Casual Factors

The following factors apparently contributed to the failure to establish containment integrity prior to achieving hot shutdown conditions during plant startup and the subsequent failure to notify the NRC:

1. A prerequisite list, that included a review of clearance logs, was rescheduled for completion from Mode 5 to Mode 2.

2. The operators performing the two recirculation spray system alignment checks noted that RS-113 was on clearance. This entry was not sufficient to flag the condition to the reviewing supervisor as a Mode 5 hold point.

3. Station personnel did not recognize the open valve as a containment integrity violation; therefore, no immediate notification was made to the Commission.

4. The mechanic who completed the work on RS-P-2A seals, failed to sign off on the clearance that the work was completed.

Technical Specification 6.8.1, Procedures and Appendix A of Regulatory Guide 1.33, requires the establishment and implementation of administrative controls and procedures for surveillance and test activities of safety-related equipment. Specific requirements are contained in Station Administrative Procedures, SAP Chapter 4 for Operations and BVPS OM Chapter 48, Conduct of Operations. Contrary to those requirements, plant startup was allowed to continue from Mode 5 to Mode 4 on December 23, 1984, without establishing containment integrity in accordance with Technical Specification 3.6.1.1. In addition to the above, inadequate review of the circumstances surrounding this event resulted in a failure to recognize that containment integrity had not been maintained as required by the limiting condition for operation contained in Technical Specification 3.6.1.1 and, as a result, a failure to notify the NRC within one hour as required by 10 CFR 50.72 and Technical Specification 6.6.1.a.

These failures to follow administrative and managerial control procedures are apparent violation(s), tracked as Unresolved Item (84-33-03).

2.5 Corrective Actions

Immediate corrective action consisted of: shutting valve RS-113; cooldown of the plant to cold shutdown conditions in order to re-establish containment vacuum; and conducting an incident critique. The licensee was evaluating long term corrective action when this special inspection was completed.

3. Introduction - Pressurizer High Level Reactor Trip Setpoint Greater Than Limiting Safety System Settings

LER: 84-018, dated January 4, 1985, reported that the actual reactor trip setpoints for the three pressurizer high level reactor trip channels were found in excess of the technical specification limit. This item was identified during the outage as a result of a system walkdown and comparison to as-built drawings undertaken by the licensee in an attempt to identify the root cause of anomalous behavior observed during plant single channel failures. Comparison of the calibration procedures with the as-built drawings indicated that an incorrect height difference between the reference leg condensate pot center line to the reference leg bellows (See attached Figure 1 from Calibration Procedure BVPS-MSP 6.41) had been used in plant procedures since initial station startup. After the correction factor was added in, it was determined that the pressurizer water level high trip setpoint would not actuate until approximately 96 percent of instrument span. Technical Specification table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, specifies a maximum allowable value of less than or equal to 93 percent of instrument span.

3.1 System Design Characteristics

The pressurizer level protection system provides three separate instrumentation channels. The high pressure instrument line tap is routed to a steam condensing pot used to generate a liquid filled reference leg. From there, the instrument line drops 7 inches before making a 90^{o} turn and is routed through the missle barrier inside containment. On the other side of the missle barrier, there is a manual isolation valve before the line branches to a pressure transmitter reference leg and a bellows sensor. The slope of this instrumentation tubing is such that an additional 6 inch drop occurs between the condensate pot and the bellows sensor. This sensor isolates the condensing pot from the high pressure (reference) line to the Barton level transmitter. A second line runs from the low pressure tap of the Barton transmitter back through the shield wall and into a lower elevation of the pressurizer.

During plant operations, there have been several occasions where a pressurizer transmitter isolation valve has experienced packing leaks which partially drained the reference leg for one instrumentation channel. When this occurred, the water between the condensing pot and the bellows sensor was lost. Routine operator checks of the three control room instrumentation channels would identify this condition as a level increase of from 5 to 7 percent on one channel. A Maintenance Work Request would be issued to return the inoperable channel to service. In each of the several instances, licensee maintenance personnel would repair the packing leak and refill the reference line with water. The affected channel would return back to a normal reading in close agreement with the two other independent channels.

Instrument and Control engineers informed the inspector that with a total water drop of 7 inches as was expected to occur when water is drained from the condensate pot to the bellows sensor, over a total span of about 200 inches on hot calibration, a total level increase of 3-1/2 percent was expected. After several occasions where the observed increase was greater, the licensee initiated an investigation during the fourth refueling outage. Calculations indicated that for a channel response of up to 7 percent full scale, the level drop between the condensate pot and bellows sensor would have to be about 13 inches. A hand-over-hand walkdown of the system identified that only the 7 inch drop between the condensate pot and the 90^{o} elbow had been taken into account in the original A-E calibration procedure. The level change due to the slope of the piping running from the condensate pot to the bellows sensor had not been included.

3.2 Technical Specification Requirement Bases

The pressurizer water level trip insures protection against reactor coolant system overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water release to the pressurizer safety valves. No credit is taken for operation of this trip in the accident analyses; however, its functional capability is required to enhance the overall reliability of the reactor protection system.

NUREG 0737, Item II.B.1, Performance Testing of Relief and Safety Valves, WOG Report on EPRI Test Program, WCAP 10105, addresses specific safety valve performance concerns (water - solid and two-phase flow conditions) uncovered during the EPRI valve testing program. This WCAP encapsulates the EPRI studies of this area, which demonstrate that water relief through the pressurizer safety valves (Target Rock) is not a significant concern.

3.3 Corrective Action

The Supervisor of Instrumentation and Control informed the inspector that revisions to the test procedures (MSP 6.41, -42, -43, Pressurizer Level Protection Channel Calibration) have been completed. Review of plant level instrumentation systems indicate that a bellows sensor system was only employed in the pressurizer level channels.

3.4 Casual Factors

All three pressurizer level channel trip setpoints were set below Technical Specification 2.2.1, Reactor Trip System Instrumentation Setpoints, allowable ranges since plant startup because of a common mode error in developing calibration procedures that did not accurately reflect elevation changes between the condensate pot and the bellows sensor. The calibration procedure development and review process did not identify this deficiency.

Technical Specification 2.2.1 requires that when a reactor trip system instrumentation setpoint is less conservative than the value shown in the Allowable Values Column of Table 2.2-1, that the channel be declared inoperable and the appropriate action statement of Specification 3.3.1.1 be applied until the channel is restored to operable status. Table 2.2-1 specifies an allowable value for the pressurizer water level high trip setpoint of less than or equal to 93 percent of instrument span. Table 3.3-1, Reactor Trip System Instrumentation, requires than when one pressurizer water level channel is inoperable, that the inoperable channel be placed in the trip position within one hour; and continued operation is permitted until the performance of the next channel functional test is required. The failure to set the three pressurizer level setpoints within the values specified is an apparent violation (85-03-01) of Technical Specification 2.2, Limiting Safety System Settings.

4. Exit Interview

The resident inspectors and a Region-based Projects Section Chief met with senior licensee management representatives on January 17, 1985, to present the inspection's scope and findings. Preliminary corrective action for two events were reviewed at that time.

L-459 PRESSURIZER LEVEL PROTECTION CHANNEL I CALIBRATION (Cont.)

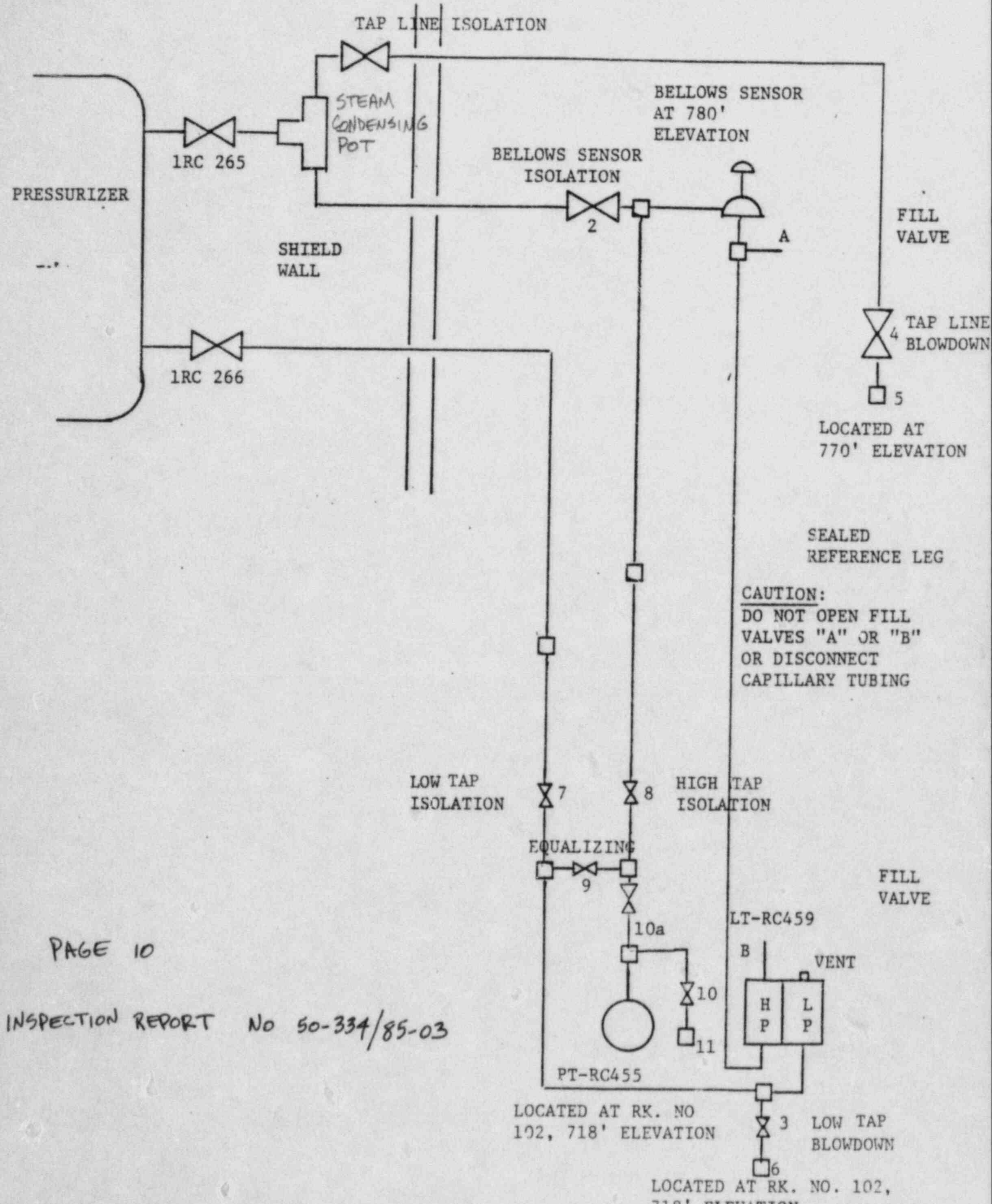

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FIGURE 1