U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-277/82-18
50-278/82-17

Docket No. 50-277
50-278

License No. DPR-44
DPR-56 Priority -- Category C

Licensee: Philadelphia Electric Company

2301 Market Street

Philadelphia, Pennsylvania

Facility Name: Peach Bottom Power Station Units 2 and 3

Inspection At: 2301 Market Street, Philadelphia, Pennsylvania

Inspection Conducted: July 29, 1982

Inspectors: P. K. Eapen
P. K. Eapen, Ph.D., Reactor Inspector — 8/23/82 date

DR Havenkamp
D. R. Havenkamp, Licensing Coordinator — 8/23/82 date

Approved By: D A Beckman
D. A. Beckman, Chief, Plant System Section — 8/23/82 date

Inspection Summary:

Inspection on July 29, 1982 (Inspection Report Nos. 50-277/82-18 and 50-278/82-17)
Areas Inspected: Routine announced safety inspection by two region-based inspectors and one supervisor of the licensee's bases for submittals to NRC regarding the availability of ECCS during DC Power Supply Failures and a request for a Technical Specification amendment for the Core Spray Sparger d/p alarm setpoint. The inspection involved 9 inspection hours at the licensee's corporate offices.
Results: No violations were identified.

DETAILS

1. Persons Contacted

W. Alden, Engineer in Charge, Nuclear Section
B. Allshouse, Engineer, Nuclear Section
W. Birely, Senior Licensing Engineer
W. Bowers, Electrical Engineer
B. Clark, Special Projects Engineer

2. Effect of DC Power Supply Failure on Emergency Core Cooling System (ECCS) Performance - Licensee's Evaluations

a. References

1) GE letter from R. E. Engel to P. S. Check (NRC) dated November 1, 1978, Subject: DC Power Source Failure for BWR/3 and 4.

2) NRC letter from T. A. Ippolito to E. G. Bauer (Philadelphia Electric Company (PECO)) dated April 25, 1980, Subject: Effect of DC Power Supply Failure on ECCS Performance.

3) PECO letter from S. L. Daltroff to T. A. Ippolito (NRC) dated August 15, 1980 (Response to Reference 2).

b. Background and Discussions

In 1978, NRC Staff identified the following concerns for Boiling Water Reactors (BWR/3's and 4's).

1) The effects of a Direct Current (DC) power source failure on ECCS performance and

2) A lack of Peak Cladding Temperature (PCT) versus break area correlation for the small piping breaks.

To address the above concerns, the General Electric Company (GE) conducted a generic study using NRC approved models of NEDO-20566 and worst-case plant configurations. GE submitted the results and conclusions based on the bounding analyses and generic study to the Commission as an attachment to Reference 1). Subsequently, in Reference 2), the NRC requested PECO to confirm the conclusions of the generic study regarding minimum ECCS equipment availability for Peach Bottom Units 2 and 3 and provide a list of ECCS equipment that would be available for large and small (1) recirculation loop discharge breaks and (2) recirculation loop suction breaks during a DC Power Supply Failure and concurrent equipment loss due to water spillage.

PECO responded to the NRC request in Reference 3) stating that the following ECCS equipment will be available in the event of a DC Power Supply Failure and loss of equipment due to water spillage:

For large and small recirculation loop discharge breaks:

Automatic depressurization system
One core spray loop (2 pumps)
One low pressure coolant injection (LPCI) pump

For large and small recirculation loop suction breaks:

Automatic depressurization system
One core spray loop (2 pumps)
Two LPCI pumps in one loop
One LPCI pump in other loop

NRC Region I has been requested to review the licensee's actions with respect to Reference 2). Purpose of this inspection is to review the licensee's bases for the conclusions stated in Reference 3).

c. Documents Reviewed

1) PECO Electrical Engineering File No. GOVT 1-1 (NRC letters) Subject: Effect of DC Power Supply Failure on ECCS Performance

2) Peach Bottom Atomic Power Station Units 2 and 3, Drawing No. E-26 (Revision 22), Sheets 1 and 2. Title: Single Line Diagrams - 125/250 VDC System Unit 2.

3) Peach Bottom Atomic Power Station Units 2 and 3, Drawing No. E-27 (Revision 12), Sheets 1 and 2. Title: Single Line Diagram - 125/250 VDC System Unit 3.

d. Details of the Review

The inspectors reviewed the documents listed in item 2.b above to establish that the information submitted by the licensee was technically adequate and satisfied the requirements of Reference 2).

The licensee presented the methods used to establish ECCS availability following a DC power supply failure concurrent with loss of equipment due to water spillage.

The licensee's study included:

1) A review of the effects of a DC power supply failure and equipment loss due to water spillage on ECCS equipment,

2) Preparation of matrix of available equipment during failures of various DC power supplies and LOCA scenarios, and

3) The effects of water spillage on ECCS equipment.

In addition to the licensee's presentation, the inspectors independently reviewed references 1), 2), and 3), including the licensee's assumptions, methods, matrix results, and failure scenarios.

e. Findings

Based on the review and presentations discussed in item 2.d, the inspectors found the licensee's submittal to be technically adequate and responsive to the requirements of Reference 2).

The inspectors have no further questions in this matter.

3. Review of the Technical Specification Change Request for the Core Spray Sparger to Reactor Pressure Vessel Differential Pressure Alarm Setpoint

a. References

1) PECO letter from E. J. Bradley to H. R. Denton (NRC) dated October 1, 1981, Subject: Technical Specification Amendment Request.

2) Inspection and Enforcement Circular (IEC) No. 79-24 dated November 26, 1979, Subject: Proper Installation and Calibration of Core Spray Pipe Break Detection Equipment on BWR's.

3) General Electric Nuclear Service Information Letter (SIL) No. 300 dated September 1979, Subject: Instrumentation for Core Spray Sparger Line Break Detection.

b. Background and Discussions

During 1976, the Iowa Electric Light and Power Company (Iowa Electric) identified and corrected a potential problem involving the Core Spray (CS) pipe break detection system at the Duane Arnold Energy Center (DAEC). The problem relates to the setpoint, function, and installation of the differential pressure (dp) instrument designed to detect a CS pipe break in the reactor vessel annulus area (i.e., located outside the core shroud but inside the reactor vessel). The installed instrument, having range of 0 - 24 psid, displayed downscale indication (i.e., reading less than zero psid) during operation.

Iowa Electric's investigation of the downscale deflection revealed that the original piping arrangement and calibration did not adequately take into account the effect of density changes of the water in the pressure leg connections. The original installation had the high pressure side of the dp instrument connected to the reference leg in the vessel and the low pressure side to the core spray piping outside the vessel but inside the drywell. With the piping intact, this arrangement senses the pressure difference between bottom and top of core. With a break in CS piping in the annulus area the instrument then senses the additional pressure drop across the separators (dp $\approx$ 7 psi additional) and dryers (dp $\approx$ 7-inches water). This installation was in accordance with GE design requirements.

Also in accordance with GE instructions the calibration of the dp instrument was performed with the reactor in the cold condition and the alarm was set to trip at 5 psid increasing. Because of this cold calibration the dp instrument then indicated full downscale negative during operation. This negative dp was due to the heat up of the reference leg which caused the fluid density to decrease as the plant reached hot conditions. The magnitude of this Δp was determined to be about 3.5 psid following completion of the modification discussed below.

Adding the 3.5 psi downscale deflection to the 5 psi alarm setpoint results in a total required deflection of 8.5 psi to initiate the alarm at the setpoint. Since the total dp available across the separators and dryer is only 7 psi (1.5 psi less than the total required deflection), the alarm would not be actuated by a CS break in the annulus. Therefore, the original installation, including calibration procedure, was deficient.

To correct the problem, Iowa Electric modified the installation by interchanging the pressure sensing connections and resetting the alarm to trip at 2 psid decreasing. In this orientation it was found that going from cold to hot reactor conditions produced a 3.5 psid positive deflection. The Technical Specifications were changed to reflect the revised alarm setpoint.

Further review by the NRC found the above lack of alarm exists on other operating BWRs.

The specific concern is that failure of the injection piping would not be detected on the plants in question because the alarm is the only control room indication involved. The actual differential pressure can only be read at local gauges located on instrument racks in the reactor building.

General Electric sent Reference 3) to utilities which recommends that:

> BWR operators, who have differential pressure (dP) instrumentation which reads only positive values, interchange their core spray line break instrument connections so that the high side connection is to the core spray sparger sensing line and the low side connection is to the above the core sensing line. This instrumentation should be calibrated for a zero dP reading during cold shutdown; it will then give a positive dP reading during normal rated power operation and a pegged zero reading for a line break indication during normal rated power operation.
>
> Also, when this change is made, the recommended alarm setpoint (on decreasing dP) setting is 0.5 $\pm$ 0.25 psid; and for those plants that have a value in their Technical Specifications, > 0 psid is recommended as a limit.
>
> This change will produce an alarm during normal shutdown. When the plant is returned to service, clearing of the alarm by a positive dP reading near rated power will indicate that the instrumentation is working.

NRC issued Reference 2) to all operating reactors and recommended that each facility review this potential problem in the following respects:

1) If the facility utilizes a core spray leak detection system similar to that described above, determine if the described problems exist. If so, initiate appropriate corrective action at the next planned refueling outage.

2) Propose changes, as appropriate, to those Technical Specifications which must be revised as a result of the above modification.

3) For interim operations until full corrective measures have been taken it is recommended that direct readings from dp gauges be periodically taken or setpoints changed along with providing necessary instructions to the operators regarding indications from this system.

In response to this circular, PECO submitted Reference 1 and proposed a change to the Core Spray Sparger to Reactor Pressure Vessel d/p Alarm Setpoint Technical Specification.

The scope of this inspection was to assure (1) the requested Technical Specification change is adequate to correct the problem identified in Reference 2, (2) the requested Technical Specification is prescribed using technically sound methods, and (3) the requested Technical Specification provides prompt indication of core spray pipe break within the stated range of uncertainty.

c. Details of Review and Discussions

The Special Projects Engineer provided a detailed discussion of PECO activities in light of References 2) and 3). These activities included:

1) A review of Peach Bottom instrumentation for Core Spray sparger line break detection,

2) Revision of ST 9.1-2, The Surveillance Log, to include readings of the core spray sparger d/p indication as an interim action pending setpoint revision, and

3) The determination of the plant specific d/p alarm setpoint for Peach Bottom Units 2 and 3.

The Special Project Engineer also demonstrated to the inspectors:

1) How the requested alarm setpoint of 1 $\pm$ 1.5 psid meets the intent of greater than zero psid limit recommended in References 2) and 3).

2) The requested Technical Specification limit of 1 $\pm$ 1.5 psid would provide an alarm to the control room in the worst case 1.2 psid before the d/p corresponding to a core spray pipe break is reached.

3) The present instrumentation at Peach Bottom Units 2 and 3 have an uncertainty of $\pm$ 1.5 psid. Therefore, choosing a Technical Specification limit 0.5 $\pm$ .25 psid would result in spurious alarms and other operational problems.

d. Findings

Based on the above presentation, the inspectors found the requested Technical Specification to be adequate and technically sound. The uncertainty band requested is unique for the instrument, cannot be reduced and is identical to that of the existing Technical Specification.

The requested change is consistent with GE SIL No. 300 and IEC-79-24 and it meets the recommended Technical Specification limit of $\geq$ 0 psid. The licensee's actions pursuant to IEC 79-24 are considered complete and the IEC closed.

The inspectors have no further questions in this matter.

4. Exit Interview

The inspectors met with the licensee's representatives denoted in Section 1 at the conclusion of the inspection to summarize the findings of the inspection as detailed in this report.

The inspectors stated:

1) Safety Evaluation Reports will be written for the items discussed in Sections 2 and 3 above and would be forwarded to NRR for issuance.

2) The inspector's conclusion was based on discussions and independent review of the PECO documents.

3) The supporting documents furnished by the licensee were part of the formal file for these matters but apparently were not maintained under PECO's quality assurance requirements for the preparation and maintenance of such documents. Preparation and upkeep of the supporting documents for licensee's sumbittals pertaining to safety-related items would be subjects for future NRC inspections (50-277/82-18-01 and 50-278/82-17-01).

The licensee's representatives acknowledged the inspector's statements.