NON-PUBLIC?: N
ACCESSION #: 8908180320
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Peach Bottom Atomic Power PAGE: 1 of 7
Station - Unit 2

DOCKET NUMBER: 05000277

TITLE: Feedwater Control Malfunction Causes Low Level Reactor Scram
EVENT DATE: 05/19/89 LER #: 89-012-01 REPORT DATE: 08/15/89

OPERATING MODE: N POWER LEVEL: 024

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: T. E. Cribbe, Regulatory Engineer TELEPHONE: (717)456-7014

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: JB COMPONENT: HS MANUFACTURER: C770
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

At 0721 on 5/19/89 with Unit 2 at 24% thermal power the FeedWater Level Control System was transferred from single element to three element control. Immediately, the "REACTOR HI-LO LEVEL" alarm annunciated, followed by both "B" and "C" Reactor Feed Pumps (RFP) tripping simultaneously. Feedwater level control was returned to single element and a restart of "C" RFP attempted. Before feed flow could be re-established, level decreased below 0 inches resulting in an automatic scram and Group II and III isolations. The "C" RFP was restarted and level decrease stopped above -48 inches. At this level, Alternate Rod Insertion backup scram initiated and both Reactor Recirculation pumps tripped. The High Pressure Coolant Injection and Reactor Core Isolation cooling systems also receive initiation signals at this level, but did not actuate as the logic was not Satisfied. Reactor water level was restored to greater than 0 inches and the unit stabilized in the hot shutdown condition. The scram and Group II and III isolations were reset and essential systems returned to service. The root cause of this event was failure of the Feedwater Level Control Selector switch. No actual Safety consequences occurred as a result of this event. The failed switch Was

replaced, feedwater control amplifiers were calibrated and procedures were enhanced. This event will be reviewed with the appropriate plant personnel. Three previous similar LERs were identified.

END OF ABSTRACT

Requirements for the Report

This LER is being submitted pursuant to 10CFR50.73(a)(2)(iv) to report those conditions which resulted in the automatic actuation of an Engineered Safety Feature (i.e., Reactor Protection System (EIIS:JC).

Unit Status at Time of Event

Unit 2 was in the Run Mode at 24% thermal power with "C" Reactor Feedwater Pump (RFP) (EIIS:P) in automatic single element control and "B" RFP running with the discharge Valve (EIIS:V) closed. Unit 3 was in the Refuel Mode with the core offloaded. There were no systems or components that were inoperable at the start of the event that contributed to the event.

Description of Event

At 0721 on 5/19/89 during normal plant startup the Feedwater Level Control system was transferred from single element (level) to three element (level, steam flow and feed flow) control. Immediately, the "REACTOR HI-LO LEVEL" alarm (EIIS:ALM) annunciated (see analysis). The Reactor Operator (RO) confirmed level as high and increasing rapidly. At this time both "B" and "C" RFPs tripped simultaneously on low suction pressure. The RO then returned the Feedwater Level Control Selector switch (EIIS:HS) to single element control and attempted to restart "C" RFP. Before feed flow could be re-established, reactor level decreased below 0 inches (172 inches above the core) causing d Reactor Protection System (RPS) automatic low level scram and Primary Containment Isolation System (PCIS) (EIIS:JM) Group II and III isolations at 0722. Isolation of the Reactor Water Cleanup system (EIIS:CE), drywell equipment and floor drain sumps (EIIS:WD), drywell and torus instrument nitrogen supply (EIIS:LK), and reactor building ventilation (EIIS:VA) occurred as a result. The "C" RFP was restarted at 0723 and level decrease stopped above -48 inches. At this level, Alternate Rod Insertion (ARI) backup scram initiated and both Reactor Recirculation pumps tripped. The High Pressure Coolant Injection (HPCI) (EIIS:BJ) and Reactor Core Isolation Cooling (RCIC) (EIIS:BN) systems also receive initiation signals at this level, but did not actuate because the required logic was not satisfied. This response was proper considering the reactor level and instrument setpoints. Reactor water level was restored to greater than 0

inches and the unit was stabilized in the hot shutdown condition at 0727. The scram and Group II and III isolations were reset and essential systems (e.g., reactor building ventilation) were returned to service at 0744.

Cause of the Event

The root cause of this event was failure of the Feedwater Level Control Selector switch (Cutler-Hammer CR2940UB203A) (see attachment 1). Upon moving the switch to the three element position, contacts 7 and 8, which close to provide the level signal input, failed to complete the circuit. Without continuity through contacts 7 and 8 the level control circuitry saw a loss of level signal. This false loss of reactor vessel level signal resulted in "C" RFP responding with a sudden increase in feedwater flow, followed by a low suction trip of both RFPs and eventually the automatic scram. Both MCGuire Unit 1 and Susquehanna Unit 1 have experienced similar failures. In these failures this particular type switch was also used in low current

(milliamps) applications whereas the switch is designed to carry up to 10 amperes. Analysis b
the manufacturer (Cutler-Hammer) determined that these failures were due to formation of silver sulfide on the silver plated contacts. The low current in this application is not sufficient to burn the silver sulfide from the contacts. Contrary to the Cutler-Hammer analysis, an analysis by the Philadelphia Electric Company Metallurgical Laboratory revealed only traces of silver sulfide, with little or none on the actual contact surface. What did exist on the contact surface was d combination of dirt and dust which resulted in the same type failure (i.e., a lack of continuity when the contacts closed). It is believed that this accumulation can be attributed to infrequent movement of this switch and the intensive level of work activity on the control panels during the prolonged shutdown.

Analysis of the Event

No actual safety consequences occurred as a result of this event.

The Reactor Protection System (RPS) and Primary Containment Isolation System (PCIS) responded properly during the reactor level transient. The HPCI and RCIC systems received initiation signals from level channels (EIIS:CHA) "A" and "B" at -48 inches, but as designed the systems did not initiate. The logic requires an "A and C", "A and D", "B and C", or "B and D" signal combination to be satisfied (see attachment 2). A review of the current surveillance test data confirmed the as found trip setpoints for channels "C" and "D" were dt a lower level than "A" and "B", and within acceptable tolerances. In summary, level did not decrease to a low enough point to

trip the required combination of channels to satisfy the Emergency Core Cooling System (ECCS) actuation logic. The ARI backup scram and Recirculation Pump trips initiated as designed at -48 inches.

During the event it was determined that the initial "REACTOR HI-LO LEVEL" alarm Was spurious followed by a valid high level condition. Investigation determined the level control selector switch to be of break before make design. This feature causes the level signal to be interrupted for a fraction of a second during switch transfer, thus initiating the alarm. The duration of circuit interruption is not sufficient to affect the control logic. The RO properly responded to the alarm, but this diverted his attention from the feedwater flow indication. Had he been aware that this alarm could be expected, his concentration would have been directed toward the faster acting feedwater flow instrumentation. Considering the short time interval between the level alarm and RFP trips it is doubtful that the outcome of this event would have differed.

The probability of this event occurring at a high power level is low as transfer to three element control is normally performed prior to exceeding 25% power. Had the event occurred dt a higher power level, level would have increased at d slower rate allowing more time for the RO to respond and mitigate the transient. If the RFPs had tripped dt a higher power level, the HPCI and RCIC systems would have been available to ensure adequate core cooling.

Corrective Actions

The following corrective actions have been taken:

1) The defective Feedwater Level Control Selector switch has been replaced with a new switch.

2) The 104 and 107 proportional amplifiers (see attachment 1) were initially recelebrated and as a precautionary measure amplifier (EIIS:AMP) 107 was replaced because of slower than normal response during dynamic testing.

3) The following procedures were revised to provide the operator with additional guidance when transferring level control modes:

a) General Procedure GP-2, Normal Plant Startup, was revised to reference the System Operating Procedure when transferring to three element control.

b) System Operating Procedure SO-6.1.A-2(3), Reactor Feedwater Automatic Level Control, was revised to include d note to advise the operator to expect the "REACTOR HI-LO" alarm when transferring the Feedwater Level Control selector switch from the single element to three element position or vice versa. A second note was added to advise the operator which instrumentation should be observed when changing feedwater control operating modes.

4) A failure analysis of the level control selector switch has been completed.

5) This event has been reviewed by licensed plant personnel.

The following corrective actions are planned:

1) The Independent Safety Engineering Group (ISEG) will complete an ISEG Event Report which will in part address Human Factors. The report will be evaluated and the appropriate recommended actions implemented.

2) Evaluate the switch failure analysis and implement those corrective actions considered necessary to prevent recurrence.

3) The existing feedwater level control system is scheduled to replaced with a digital "fault tolerant" system (Modification 1843) during the next Unit 2 and Unit 3 refueling Outages. This system is designed to prevent feedwater transients due to single failures.

4) Unit 3 Feedwater Level Control Selector switch will be removed, tested and if necessary replaced prior to Unit 3 restart.

Previous Similar Events

There were three previous similar LERs associated with the failure of electrical components in the Feedwater Level Control System and resulted in a low reactor water level scram. The following is a brief description of the failure and those actions taken to prevent recurrence:

1) LER 3-84-11 was believed to be an intermittent component failure in the feedwater control system. Those components which could have produced a similar failure were replaced.

2) LER 2-85-22 involved a loose connection on the output of the

electronic feedwater flow summer circuit. The summer circuit and connector were replaced.

3) LER 3-86-20 involved corrosion at the terminal connections for a temperature element which provides a feedwater density compensation signal to the feedwater control system. The affected terminals were cleaned and output monitored until it was certain that the problem was corrected. The digital "fault tolerant" system scheduled to be installed during the next Unit 2 and 3 refueling outages was identified in this LER as being planned for future installation.

Based on the apparent failure mechanism of the Feedwater Level Control Selector Switch, the corrective actions taken after the previous similar events would not have prevented this event. The corrective actions either addressed a specific component or a future modification not yet implemented.

TEXT PAGE 6 OF 7

Figure "Feedwater Level Control, Attachment #1" omitted.

TEXT PAGE 7 OF 7

Figure "ECCS Logic at - 48" Reactor Water Level, Attachment #2" omitted.

ATTACHMENT 1 TO 8908180320 PAGE 1 OF 1

CCN-89-14125

PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM ATOMIC POWER STATION
R. D. 1, Box 208
Delta, Pennsylvania 17314
(717) 456-7014

PEACH BOTTOM-THE POWER OF EXCELLENCE

D.M. Smith
Vice President

August 14, 1989

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SUBJECT: Licensee Event Report
Peach Bottom Atomic Power Station - Unit 2

The revised LER concerns providing the Component failure information (item 13) for the Feedwater Level Control Selector Switch identified in the event. This revision also updates the corrective action section.

Reference: Docket No. 50-277
Report Number: 2-89-012
Revision Number: 01
Event Date: 05-19-89
Report Date: 08-15-89
Facility: Peach Bottom Atomic Power Station
RD 1, Box 208A, Delta, PA 17314

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv).

Sincerely,

cc: T. P. Johnson, USNRC Senior Resident Inspector
W. T. Russell, USNRC, Region I

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