10 CFR 50.54(f)
GL 89-19

PHILADELPHIA ELECTRIC COMPANY

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DAVID R. HELWIG
VICE PRESIDENT
NUCLEAR SERVICES

March 20, 1990

Docket Nos. 50-277
50-278
50-352
50-353

License Nos. DPR-44
DPR-56
NPF-39
NPF-85

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Peach Bottom Atomic Power Station, Units 2 and 3
Limerick Generating Station, Units 1 and 2
Response to NRC Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47 Safety Implication of Control Systems in LWR Nuclear Power Plants"

REFERENCES: (1) Letter from S. D. Floyd (BWROG) to J. G. Partlow (NRC), dated February 16, 1990

(2) Letter from J. G. Partlow (NRC) to S. D. Floyd (BWROG), dated March 20, 1990

Gentlemen:

NRC Generic Letter 89-19 dated September 20, 1989, required licensees to provide, within 180 days of the date of the Generic Letter, a statement as to whether licensees will implement the recommendations provided in Enclosure 2 of the Generic Letter. The Generic Letter also required licensees to provide a schedule for implementation of the recommendations and the basis for the schedule or to provide appropriate justification if the recommendations were not going to be implemented.

Attached is the Philadelphia Electric Company (PECo) response to the Generic Letter for Peach Bottom Atomic Power Station

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(PBAPS), Units 2, and 3. The NRC recommendations are restated followed by PECO's response.

As outlined in Reference 1, the Boiling Water Reactor Owners' Group (BWROG) is addressing this issue on a generic basis. Based on our understanding of this work to date, we anticipate that the results will be applicable to Limerick Generating Station (LGS) Units 1 and 2. PECo plans to exercise the 45-day extension for submittal of a response as approved by the NRC in Reference 2. Accordingly, the response for LGS Units 1 and 2 will be provided by May 4, 1990.

If you have any questions, or require additional information, please contact us.

Very truly yours,

DRHelwig

Attachments

cc: W. T. Russell, Administrator, Region I, USNRC
J. J. Lyash, USNRC Senior Resident Inspector, PBAPS
T. J. Kenny, USNRC Senior Resident Inspector, LGS

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

D. R. Helwig, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company, that he has read the response to Generic Letter No. 89-19, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

DR Helwig

Vice President

Subscribed and sworn to before me this 20th day of March 1990.

Catherine A. Mendez

Notary Public

NOTARIAL SEAL
CATHERINE A. MENDEZ, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires Sept. 4, 1993

Peach Bottom Atomic Power Station, Units 2 and 3
Response to Generic Letter 89-19
"Request for Action Related to Resolution of
Unresolved Safety Issue A-47 Safety Implication
of Control Systems in LWR Nuclear Power Plants"

NRC Recommendation (1)(a)

It is recommended that all GE boiling-water-reactor (BWR) plant designs provide automatic reactor vessel overfill protection to mitigate main feedwater (MFW) overfeed events. The design for the overfill-protection system should be sufficiently separate from the MFW control system to ensure that the MFW pump will trip on a reactor high-water-level signal when required, even if a loss of power, a loss of ventilation, or a fire in the control portion of the MFW control system should occur. Common-mode failures that could disable overfill protection and the feedwater control system, but would still result in a feedwater pump trip, are considered acceptable failure modes.

It is recommended that plant designs with no automatic reactor vessel overfill protection be upgraded by providing a commercial-grade (or better) MFW isolation system actuated from at least a 1-out-of-1 reactor vessel high-water-level system, or justify the design on some defined basis.

PECo Response

The existing PBAPS Reactor Feedwater Pump Turbine (RFPT) high level trip logic uses the feedwater control system to detect reactor vessel high level. There is redundant high level detection provided by the High Pressure Coolant Injection (HPCI) System. This redundant HPCI high level trip provides adequate power supply independence, sensing line independence, separation of sensors, physical separation, indication and alarming from the Feedwater Reactor Level Control System. Details supporting this conclusion are provided in the attached Design Analysis (Attachment 2 to this Generic Letter response). Therefore, it is concluded that the current design meets the recommendations of the Generic Letter, and no plant modifications are necessary.

NRC Recommendation (1)(a) Continued

In addition, it is recommended that all plants reassess their operating procedures and operator training and modify them if necessary to ensure that the operators can mitigate reactor vessel overfill events that may occur via the condensate booster pumps during reduced pressure operation of the system.

PECo Response

Peach Bottom Atomic Power Station does not have condensate booster pumps; however, the multi-stage condensate pumps installed at PBAPS produce sufficient discharge pressure and capacity such that a booster pump is not required. Our assessment assumes that the PBAPS condensate pumps perform similarly to condensate booster pumps.

Reduced pressure operation, as it is discussed in the Generic Letter, is when Reactor Pressure Vessel (RPV) pressure is less than the shutoff head of the condensate pump, and, therefore, the condensate pumps can pump directly into the Reactor Pressure Vessel. At PBAPS, the condensate pumps have a shutoff head of approximately 600 psi. Pressures of less than 600 psi in the reactor vessel can only occur during the following three scenarios: 1) during a major plant transient, 2) startup or 3) shutdown. For each of these scenarios, PECo evaluated whether current operating procedures and operator training were adequate to ensure that operators can prevent and mitigate reactor vessel overfill events due to condensate pumps during reduced pressure operation. Each scenario is addressed below:

Plant Transient

The post scram scenario presents the most plausible condensate pump overfill possibility of the three scenarios in which condensate pumps are operated below 600 psi. In numerous industry scrams and several PBAPS scrams, pressure has dropped below 600 psi during the transient. The following two operating experience documents were previously reviewed by PECo and detail several overfill events:

1. Information Notice (INN) 88-77: Inadvertent Reactor Vessel Overfill

2. Institute of Nuclear Power Operations Significant Event Report (SER) 5-88: Flooding Main Steam due to Improper Reactor Water Level Control

In response to these documents, PECo identified and implemented the following corrective actions:

1. Revised existing Simulator Exercise Guides (SEG's) to incorporate some of the lessons learned from SER 5-88 and included a discussion of the possible consequences of filling the main steam lines with water;

2. Included extensive training on level control during plant transients and in particular, during scram recovery in the six week simulator continuing training cycle that started 10/10/88;

3. Developed a Licensed Operator lesson plan (LOR 88-09A) for training the operators on various industry overfill events; and

4. Committed to revise the trip procedures to provide adequate direction to prevent vessel overfill from low pressure injection systems.

In January of 1989, Operations Section Performance Standard (OSPS)-4 was issued to evaluate Reactor Operator performance during a scram. OSPS-4 requires the operator to maintain level between 0 inches and 45 inches in order to receive an "excellent rating". An "unsatisfactory rating" would result if RPV level reached 60 inches. Programmatically, Reactor Operators are trained to trip the 'A' and 'B' RFPTs when level starts to recover following the scram. Level control is then established via the 'C' RFP discharge valve bypass level control valve. The feedpump discharge valves are subsequently shut. With the discharge valves shut, the only flow path to the vessel is through the 10 inch bypass valve. The rate of level increases due to condensate pumps is strictly limited. If level should increase, the following actions would sequentially occur.

a) The 'C' RFPT will auto trip at 45 inches (Note, the RPV high-low level alarm will already be present due to the level transient)

b) In accordance with procedure OT-110, the operator will verify all RFPTs are tripped, HPCI is tripped, and the RCIC steam supply valve is shut.

c) If water level reaches 90 inches, the operator will close the MSIVs.

Procedure OT-110 will be revised by 7/20/90 to include specific guidance to the operator on what action to take on high water level if RPV pressure has decreased below condensate pump shutoff pressure.

Startup

During plant startup, one condensate pump is used to pump water from the hotwell back to the hotwell while RPV level is being maintained by matching Control Rod Drive (CRD) flow in with Reactor Water Cleanup (RWCU) flow being rejected (Step 5.16 of procedure GP-2, "Normal Plant Startup"). If additional feed is required to maintain level in the proper operating range, then makeup is provided by throttling startup recirculation valve MO-2(3)663 as necessary to raise pressure to inject condensate into the RPV. At 150 psi, RPV level control is maintained by valve AO-8(9)091 (Step 6.13 of procedure GP-2). Valve A0-8(9)091 is a ten inch bypass around the 'C' Reactor Feedwater Pump (RFP) discharge valve installed specifically for startup level control. It is controlled by the normal feedwater level control system and will automatically control level in the normal band.

A failure of the feedwater level control system could result in level increase or decrease depending on the type of failure. However, an increase in level would be detected by the operator by either visual indication on the level recorder or by a high level alarm at 29 inches (Main Steam Lines are at 110 inches). Procedure OT-110, Reactor High Level, currently is entered only when a high level alarm is received with the mode switch in "RUN". In addition, there is no clear direction if the cause of the high water level is from the condensate pumps. Therefore, OT-110 will be revised by 7/20/90 to provide guidance for overfill resulting from the condensate pumps, and to allow the procedure to be used in other modes whenever feedwater level control is in automatic. The procedure will also contain specific guidance concerning a high level condition during reduced RPV pressure conditions when condensate pumps can feed the vessel. This will ensure the operators are able to mitigate a RPV overfill event from condensate pumps during startup conditions with a feedwater level control system component failure.

Shutdown

Procedure GP-3, "Normal Plant Shutdown", is the procedure used to shutdown the plant. The plant Operations group has determined through review that the possibility of an overfill event due to reduced pressure operation is very remote. In addition, shutdowns generally use a planned reactor scram at about 30% power, at which time the Trip Procedures are entered. As previously mentioned in the post scram discussion, the planned procedure changes, along with current operator training, will adequately address this concern.

To summarize, current operator training adequately addresses the issue of RPV overfill due to condensate pumps during reduced RPV pressure operation. The following revisions to procedures will be made to ensure that operators can mitigate reactor vessel overfill events that may occur during reduced pressure operation.

a) OT-110 will be revised by 7/20/90

b) Trip procedures will be revised by 7/31/90 as part of the Rev. 4 upgrade

NRC Recommendation (1)(b)

It is recommended that plant procedures and Technical Specif[illegible] all BWR plants with main feedwater overfill rote[illegible] provisions to verify periodically the operability [illegible] ion and ensure that automatic overfill protection to mitigate main feedwater overfeed events is operable during power operation. The instrumentation should be demonstrated to be operable by the performance of a channel check, channel functional testing, and channel calibration, including setpoint verification. The Technical Specifications should include appropriate limiting conditions for operation (LCOs). These Technical Specifications should be commensurate with the requirements of existing plant Technical Specifications for channels that initiate protective actions. Previously approved Technical Specifications for surveillance intervals and limiting conditions for operation (LCOs) for overfill protection are considered acceptable.

PECo Response

As discussed in the attached Design Analysis, the Reactor Feedwater Pump Turbines can be tripped on reactor high level by either 1) activation of Feedwater Level Control relay 6A-K1 OR 2) actuation of HPCI relays 23A-K52 and 23A-K37. Presently, these specific relays and their associated contacts are not adequately addressed by Technical Specifications (TS). Therefore, an appropriate Technical Specification Change Request (TSCR) will be submitted to the NRC by January 31, 1991. Limited requirements for reactor level indication already exist in TS Tables 3.2.F and 4.2.F. Any necessary revisions to these requirements to conform with the recommendations of the Generic Letter will be included in the TSCR discussed previously. The frequency of the channel check and channel calibration will be consistent with other channels that initiate protective actions. However, it is proposed that the frequency of the channel functional testing will be once per operating cycle. This is less frequent than the current TS frequency of Logic System Functional Tests (LSFTs) for other channels that initiate protective actions. Other LSFTs are on a once per six month or once per three month frequency. PECo is considering a TSCR to extend these LSFTs to a once per operating cycle frequency. The justification for extended channel functional testing frequency is that testing relays requires temporary circuit alterations (i.e., lifted leads, jumpers and booted relay contacts). Each circuit alteration has an associated risk and possible error that could result in circuit component damage or ultimately a reactor scram if performed during power operation. Therefore, the proposed channel functional testing for vessel overfill protection provided by HPCI and Feedwater Level Control is consistent with the planned frequency of other channels which initiate protective actions.

As recommended on page 2 of the Generic Letter, short-term measures will be taken to provide periodic verification and testing of the overfill protection system. Appropriate plant procedures will be developed or revised to implement the channel check, channel functional testing and channel calibration as discussed above. This will be completed by prior to startup from the next refueling outage of Unit 2 (between cycles 8 and 9).

NUCLEAR ENGINEERING DEPARTMENT

DESIGN ANALYSIS COVER SHEET

M-20594 Rev. 12/89
DOCTYPE 152

MOD# EWR #P-51490

MOD DESCRIPTION: NRC Generic Letter 89-19--Request for action related to resolution of unresolved safety issue A-47, "Safety Implication of Control Systems in LWR Nuclear Power Plants" pursuart to 10CFR50.54(F)

STATION: PBAPS UNIT: 2 & 3

PURPOSE OF ANALYSIS: To review the Feedwater Control System to determine if the NRC concerns raised in Generic Letter 89-19 are valid for Peach Bottom APS Units 2 and 3.

ORGANIZATION	RESPONSIBLE ENGINEER	DATE	INDEPENDENT REVIEWER	DATE	DEPTH OF REVIEW: EXHIBIT 34-II	DEPTH OF REVIEW: SEE DOCUMENTATION
ORIGINATING ORGANIZATION	T. J. Cabrey	3/20/91	[signature]	3/20/91	✓	
INTERFACING BRANCH	N/A					
INTERFACING BRANCH	N/A					

BRANCH HEAD - ORIGINATING ORGANIZATION: R. C. Welsh DATE: 3/20/91

COPY TO: DAC (61B-5)
T. J. Cabrey (63A-5)
M. C. Kray

REV. NO. 2

PAGE 1 OF 10

EXHIBIT 3.4-IV
1/90

[illegible]TION

Generic Letter 89-19 (reference 1) requests action related to the resolution of unresolved safety issue A-47, "Safety Implication of Control Systems in LWR Nuclear Power Plants". In this Generic Letter the NRC concluded that protection should be provided for certain control system failures. In particular, protection should be provided for reactor vessel overfill. The purpose of this design analysis is to determine if the feedwater control system provides adequate protection to prevent an overfill event from the feedwater control system. Specific features to be analyzed are: power supply interdependence, sharing of sensors between control and trip logic, separation of control and trip circuits, and designs for indication and alarm.

PURPOSE

The purpose of this design analysis is the review of the feedwater control system to determine if the NRC concerns raised in Generic Letter 89-19 are valid for Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3.

REFERENCES

1. Generic Letter 89-19 — Request For Action Related To Resolution Of Unresolved Safety Issue A-47 "Safety Implication Of Control Systems In LWR Nuclear Power Plants" Pursuant To 10CFR50.54(f).

2. NUREG-1217 — Evaluation Of Safety Implications Of Control Systems In LWR Nuclear Power Plants - Technical Findings Related To USI A-47.

3. 6280-E-129 Rev 25 — Electrical Schematic Diagram - Reactor Feed Pump Control Scheme.

4. 6280-M1-S-25 Rev 46 — Elementary Diagram - Feedwater Control System.

5. 6280-M1-S-36 Rev 62 — Elementary Diagram - High Pressure Coolant Injection System.

6. 6280-M1-S-65 Rev 86 — Elementary Diagram - Residual Heat Removal System.

7. 6280-M-352 Rev 32 — P&ID - Nuclear Boiler Vessel Instrumentation.

8. 6280-E-28 Rev 43 — Single Line Diagram - Instrumentation and Uninterruptible AC Systems Unit 2 and Common.

9. 6280-E-29 Rev 33 — Single Line Diagram - Instrumentation and Uninterruptible AC Systems Unit 3.

10. 6280-E-3031 Rev 2 — Electrical Schematic Diagram - Panels 20C818 and 20C819 Power Distribution

DESIGN REVIEW

Refer to Figure #1 for a composite schematic of the Reactor Feedwater Pump Turbines (RFPTs) trip logic.

TRIP LOGIC

The reactor high level trip of the RFPTs comes from the 6A-K1, 23A-K52, or 23A-K37 relays. These relays are arranged such that the 6A-K1 relay by itself or the 23A-K52 AND 23A-K37 relays can trip the RFPTs on high reactor level. All of these relays use NORMALLY OPEN contacts that close to cause a trip signal. Upon closing to complete the circuit the main trip solenoid valve (SV-12) for each RFPT energizes resulting in the RFPTs tripping.

The 6A-K1 relay is controlled from alarm units 6-121B and 6-121C. The alarm units use NORMALLY CLOSED contacts which close on reactor high level. These alarm units are arranged such that both alarm units must trip (on high level) in order to energize the 6A-K1 relay which in turn will energize the SV-12s trip solenoids. These alarm units receive their signal from the reactor level pressure compensation proportional amplifiers (LPAM-6-72s). These proportional amplifiers take a reactor level delta-pressure signal (LT-6-52) and adjusts this delta-pressure signal with a reactor pressure signal (PT-6-53) to account for changes in water density. The output of the proportional amplifiers is pressure compensated reactor level. This pressure compensated level signal is then used as the controlled variable for the feedwater control system.

The 23A-K37 and 23A-K52 relays are controlled from the Foxboro reactor water level compensation panels (C818 and C819). These panels consist of Input/Output cards and a micro-processor (XAM-2-3-117s). The micro-processor receives inputs from wide range (LT-2-3-72s) and fuel zone (LT-2-3-73s) reactor level delta-pressure transmitters and a reactor pressure (PT-2-3-404s) transmitter. The micro-processor then performs pressure compensation on the two reactor level delta-pressure signals. The micro-processors also performs alarming and trip functions on these pressure compensated level signals. One of these trip functions is the reactor high level trip signal (XS-2-3-115s) to the 23A-K37 relays (similar for 23A-K52). The trip contact uses a NORMALLY OPEN contact which closes on a high reactor level. Other outputs of the mirco-processor are used for indication (control room indicators and recorders) and Emergency Core Cooling System (ECCS) reactor level and pressure trip signals.

POWER SOURCES

Table #1 summarizes all the different systems and sub-systems power requirements.

The RFPT trip logic is powered from 125 VDC and requires power to be present in order to remotely trip the RFPTs. (Manual trip can always be performed locally at the RFPT front standard.) The trip solenoid (SV-12) energizes to trip.

The feedwater control system is powered from Uninterruptable Power Supply (UPS). This UPS system receives power from either 480 VAC (E12(3)4-R-C [2(3)OB36]) or from 125 VDC (Bus 'D'). Either of these power sources being available will make continuous power available to the UPS and hence the feedwater control system. A partial (individual fuse or circuit breaker) or complete loss of UPS power to the feedwater control system could result in any of the following:

1. No impact to control system.

2. Partial loss of feedwater control system functions. This could result in the lockup (loss of control signal) of one or two RFPTs with one remaining available for control of reactor level. One RFPT for control is sufficient to control reactor level within the range of the high and low level alarms.

3. A complete loss of all three (3) RFPTs for control (all RFPTs lockup). With all three RFPTs locked-up the fixed rate of feedwater make-up to the reactor could be either less than or greater than the rate which is required to maintain reactor level steady-state. In this case level would either decrease or increase depending on when the lockup occurred.

4. If the power source to the GA- [illegible] ay coil was lost the feedwater control system could no [illegible] vide any reactor high water level trip signals to the RFPT trip logic.

The High Pressure Coolant Injection (HPCI) logic is powered from 125 VDC ('A' and 'B' Buses). A loss of either of these power sources will result in the inability of the HPCI system to provide a complete (2-out-of-2) high level trip signal to the RFPT trip logic.

The reactor water level compensation system (C818 and C819 panels) are powered from a dual power supply (E/S-2-3-122s). These dual power supplies are powered from two independent sources. The C818 panel is powered from 120 VAC (2[3]OY35) and 125 VDC (Bus 'C'), whereas the C819 panel is powered from 120 VAC (OOY03) and 125 VDC (Bus 'D'). Either power source being available to one of these dual power supplies is sufficient to power its associated panel (C818 or C819). Loss of both power sources to one of these dual power supplies will result in the associated panels from being unable to provide a high reactor level trip signal to the HPCI system and hence the RFPT trip logic.

TABLE #1
POWER SOURCES ASSOCIATED WITH THE
RFPT CONTROL AND HIGH LEVEL TRIP LOGIC

System/Subsystem	Power source	Comments
RFPT Trip Logic (SV-12A,B,C)	125 VDC Bus A,B,C	1. Requires power to remotely trip RFPTs
Feedwater Controls	120 VAC UPS 2(3)0Y50	1. Requires power to control reactor level. 2. Loss of power to this portion results in High level trip signal from the feedwater control system and lockup of one or more of the RFPTs
Feedwater Logic (6A-K1)	120 VAC UPS 2(3)0Y50	1. Loss of this portion of UPS power results in the inability to provide RFPTs high level trip signal from the 6A-K1 relay.
HPCI Logic (23A-K37)	125 VDC Bus 'A'	1. Loss of power results in the inability to provide RFPTs high level trip signal from the 23A-K37 relay.
HPCI Logic (23A-K52)	125 VDC Bus 'B'	1. Loss of power results in the inability to provide RFPTs high level trip signal from the 23A-K52 relay.
Reactor Level (XS-2-3-115C)	120 VAC 2(3)0Y35 OR 125 VDC Bus 'C'	1. Power is provided via a dual power supply where either power input available will maintain power to the components. 2. Loss of both power sources will result in the inability to provide RFPTs high level trip signal from XS-2-3-115C.
Reactor Level (XS-2-3-115D)	120 VAC 00Y03 OR 125 VDC Bus 'D'	1. Power is provided via a dual power supply where either power input available will maintain power to the components. 2. Loss of both power sources will result in the inability to provide RFPTs high level trip signal from XS-2-3-115D.

PHYSICAL SEPARATION

The RFPT trip solenoid valve (SV-12) is located near the Reactor Feedwater Pump (RFP) bays on the turbine deck, 135′ elevation. The high level trip logic circuit is power from the C06C panel in panels (C18, C32, and C39) located in the cable spreading room. From the cable spreading room the circuit routes out to the RFP bays (A,B,CC124).

The majority of the feedwater control circuitry resides in the cable spreading room in panel C18. The balance of the feedwater control circuitry resides in the main control room panel (C06C).

The feedwater control system receives its reactor and level signals from local field transmitters (LT-6-52s and PT-6-53s) which are located in the reactor building on instrument racks C65A and C65B. The feedwater control logic contact output is used in the RFPT trip logic.

The HPCI logic is located in the cable spreading room in panels C32 and C39. The HPCI logic receives its input from two other panels also located in the cable spreading room, C818 and C819. The HPCI logic output contacts are used in the RFPT trip logic.

The reactor water level panels (C818 and C819) reside in the cable spreading room. The reactor water level instrumentation receives its reactor level and pressure signals from local field transmitters (LT-2-3-72s, LT-2-3-73s, and PT-2-3-404s) which are located in the reactor building on instrument racks C65A, C65B, C91A, and C91B. Power for the reactor water level instrumentation comes from two other cable spreading room panels (C722A and C722B). Contact outputs from the reactor water level instrumentation are used in the HPCI logic in panels C32 and C39.

ARRANGEMENT OF SENSORS

Figure #3 shows the arrangement of the level transmitters which are associated with the reactor high level trip logic. One common instrument penetration (N12A) supplies steam to two (2) condensing chambers (2A and 3A). These condensing chambers condense steam into condensate to maintain a constant head of water to the level transmitters (LT-2-3-72C and LT-6-52C) reference ports. Another instrument penetration (N11A) provides a variable pressure signal to LT-6-52C. A separate instrument penetration (N16A) provides a different variable pressure signal to LT-2-3-72C. These variable pressure signals consist of reactor steam dome pressure and the head of water above the instrument penetration (variable dependant upon reactor level). This arrangement is then duplicated for the LT-2-3-72D and LT-6-52B transmitters. Reactor level is inversely proportional to the differential pressure across the transmitter. That is, for a minimum differential pressure (P(ref) = P(var)) level is at its maximum value and for a maximum differential pressure (P(ref) >>> P(var)) level is at its minimum value.

ANALYSIS

POWER SUPPLY INTERDEPENDENCE

A single power failure that could result in the feedwater control system causing reactor level to rise would be a complete failure of the 120 VAC UPS power supply (assuming the RFPTs all lockup in such a way to result in a continuous net vessel level increase). The loss of 120 VAC UPS also results in the inability of the feedwater control system to provide a high reactor level trip signal to the RFPTs trip logic via the 6A-K1 relay. However, the reactor water level system and HPCI logic would remain available to provide the reactor high level trip signal to the RFPTs to stop the vessel level rise. Note that this transient would be a relatively slow transient compared to others discussed later on in this analysis.

A loss of the 125 VDC Bus 'A' power source would result in the inability to remotely trip the 'A' RFPT ('A' SV-12 powered from 125 VDC Bus 'A') and the inability of the HPCI logic to provide a high level trip (23A-K37 logic powered from 125 VDC Bus 'A'). However, loss of this bus will have no effect on the feedwater control systems ability to control reactor level. In addition, the feedwater control system would still be able to provide a high level trip signal to trip the 'B' and 'C' RFPTs.

A loss of the 125 VDC Bus 'B' power source would result in the inability to remotely trip the 'B' RFPT ('B' SV-12 powered from 125 VDC Bus 'B') and the inability of the HPCI logic to provide a high level trip (23A-K52 logic powered from 125 VDC Bus 'B'). However, loss of this bus will have no effect on the feedwater control systems ability to control reactor level. In addition, the feedwater control system would still be able to provide a high level trip signal to trip the 'A' and 'C' RFPTs.

A loss of the 125 VDC Bus 'C' power source would result in the inability to remotely trip the 'C' RFPT ('C' SV-12 powered from 125 VDC Bus 'C'). There are no other effects on either the control circuits or trip logics as a result of this loss of power. The only other system that receives power from this bus is the 'C' reactor water level instrumentation. However, since the reactor water level instrumentation is powered by a dual power supply. Loss of this power feed doesn't have any effect on the operation of this instrument.

A loss of any one of the 120 VAC 00Y03 or 2(3)0Y35 or the 125 VDC Bus 'D' power sources does not have any effect on either the control circuits or trip logic. The reactor water level instrumentation is powered by dual power supplies. Loss of any single power feed to these power supplies doesn't have any effect on the operation of the instrumentation.

A loss of both feeds (i.e. as a result of a fire in the C722A panel) to either one of these dual power supplies will result in that channel of the reactor water level instrumentation inability to provide a high level trip signal to the HPCI logic. The feedwater control system will not be affected by this power loss. Therefore, the feedwater control system will continue to maintain level control and have the ability to provide a high level trip signal if required. 2

There are no other single power failures that by themselves would result in the feedwater control system causing reactor vessel level to rise.

SENSING LINE INTERDEPENDENCE

A failure of a sensing line to any level transmitter could result in a false level indication. This section looks at single failures of sensing lines associated with the reactor high level trip logic. This section only looks at single failures for half of the sensing lines since the other half would have similiar results (refer to Figure #3). Failures can range from small leaks to complete breaks of the sensing line.

1. Vessel Penetration N12A to Condensing Chambers 2A and 3A:

   A failure of this line will result in P(ref) decreasing which results in a false high level indication from LT-2-3-72C and LT-6-52C. If the feedwater control system is controlling on the LT-6-52C the feedwater control system would reduce feedwater flow in response to this false high level indication. If the failure was large enough the false high level indication could cause a false high level channel trip signal from LT-2-3-72C and LT-6-52C. (Note this does not result in a complete trip signal).

2. Condensing Chamber 2A to Level Transmitter LT-2-3-72C:

   A failure of this line will result in a false high level indication from LT-2-3-72C only. This failure will have no effect on the feedwater control system. If the failure was large enough the false indication could cause a false high level channel trip signal from LT-2-3-72C. (Note this does not result in a complete trip signal). If the failure was large enough so that the leak becomes uncovered the sensing line from penetration N12A to the condensing chambers 2A and 3A would also depressurize. This failure would be covered by (1) above. 2

3. Condensing Chamber 3A to Level Transmitter LT-6-52C:

   This failure is similiar to the failure of the condensing chamber 2A to level transmitter LT-2-3-72C in (2) above. However, for this failure if the feedwater control system is controlling on LT-6-52C the feedwater control system would reduce feedwater flow in response to a false high level single.

4. Vessel Penetration N11A to Level Transmitter LT-6-52C:

   A failure of this line will result in P(var) decreasing which results in a false low level indication from LT-6-52C. If the feedwater control system is controlling on LT-6-52C the feedwater control system would increase feedwater flow and actual level would rise. If actual reactor level reaches the high level trip setpoint the feedwater control system will not be able to initiate a high level trip signal. However, the HPCI system will initiate a high level trip signal via the LT-2-3-72C and LT-2-3-72D transmitters.

5. Vessel Penetration N16A to Level Transmitter LT-2-3-72C:

   This failure is similiar to the failure for vessel penetration N11A to level transmitter LT-6-52C in (4) above except the feedwater control system will not be affected. 2

Therefore, there is no potential for reactor overfill from any single failure of the sensing lines associated with the reactor high level trip logic.

SHARING OF SENSORS

The feedwater control system does share sensors and compensation instrumentation between control and reactor high level trip to the RFPTs functions. However, there is a redundant reactor high level trip signal available from the HPCI logic. The redundant trip signals come from diverse and independent sensors and pressure compensation instrumentation.

SEPARATION OF FUNCTIONS

The feedwater control system does share compensation instrumentation between control and high level trip functions. 2 Therefore, there is no separation within the feedwater control system between the control and trip functions. The panels that contain the instrumentation that makes up the redundant trip signals reside in Panels C32, C39, C818, and C819. The C18 panel contains the feedwater control system. Therefore, a fire in any one of the above listed panels will only affect one portion of the high level trip logic (feedwater or redundant trip signal). 2 The remaining trip channel will be available to provide a reactor high level trip signal if required. This arrangement provides sufficient separation between the redundant trip signals and the reactor level control functions.

LOSS OF VENTILATION

The panels that house the majority of the reactor high level trip logic and the feedwater control system reside in the plant cable spreading room. Normally the ventilation system maintains the cable spreading room at approximately 85°F. The equipment contained in this room is designed to operate at temperatures up to 120°F. Loss of ventilation to the cable spreading room will result in a control room trouble alarm and a slow temperature rise. Operator action would then be taken to restore normal ventilation. At 110°F any one of 5 EHC cabinets temperature switch would alarm (the EHC cabinets are directly across an aisle from the feedwater control panel). At this point additional operator action would be taken. Based on the above it is expected that appropriate operator action can be taken before the cable spreading room temperature exceeds 120°F.

INDICATION AND ALARMS

Both the feedwater control system and the reactor water level instrumentation system provide control room indication of reactor level. Both systems provide indication on indicators and trend recorders. The feedwater control system reactor water level indication ranges from 0" to +60". The reactor water level instrumentation system indication ranges from -325" to +60". The RFPT high reactor level trip setpoint is at +45".

Both the feedwater control system and the HPCI logic provide high reactor level trip alarms in the control room. In addition, the feedwater control system provides a reactor high level alarm at 29".

MOD 1843 IMPACT

Modification 1843 replaces the feedwater control system with a smart, fault tolerant, Digital Feedwater Control System (DFCS). The DFCS will replace the high reactor level trip logic to the 6A-K1 relay. The new logic will require both DFCS computers to call for a trip or be off line taken twice (see figure 2). The power to the 6A-K1 relay is not effected by this modification. This modification has no effect on the redundant reactor high level trip signal provided by the HPCI and reactor water level instrumentation systems.

CONCLUSION

Based on the above analysis the Peach Bottom APS Units 2 and 3 meet the requirements of NRC Generic Letter 89-19. Specifically, there is adequate power supply independence, sensing line independence, separation of sensors, physical separation, indication, and alarming between the feedwater reactor level control system and the RFPTs high reactor level trip logics.

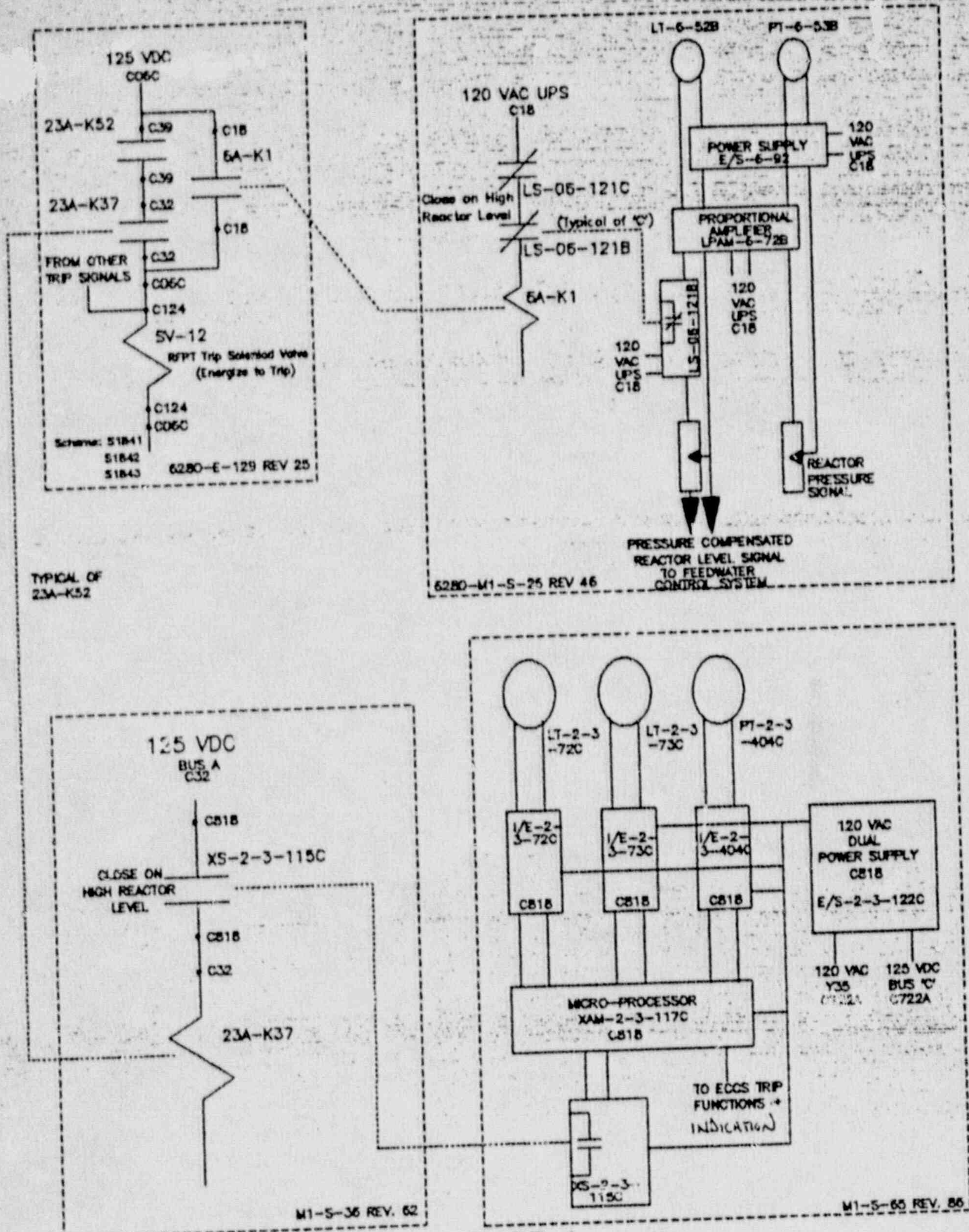

FIGURE 1
GENERIC LETTER 89-19 DESIGN ANALYSIS
COMPOSITE SCHEMATIC OF RFPT HIGH LEVEL TRIP LOGIC

TJC 1/24/90

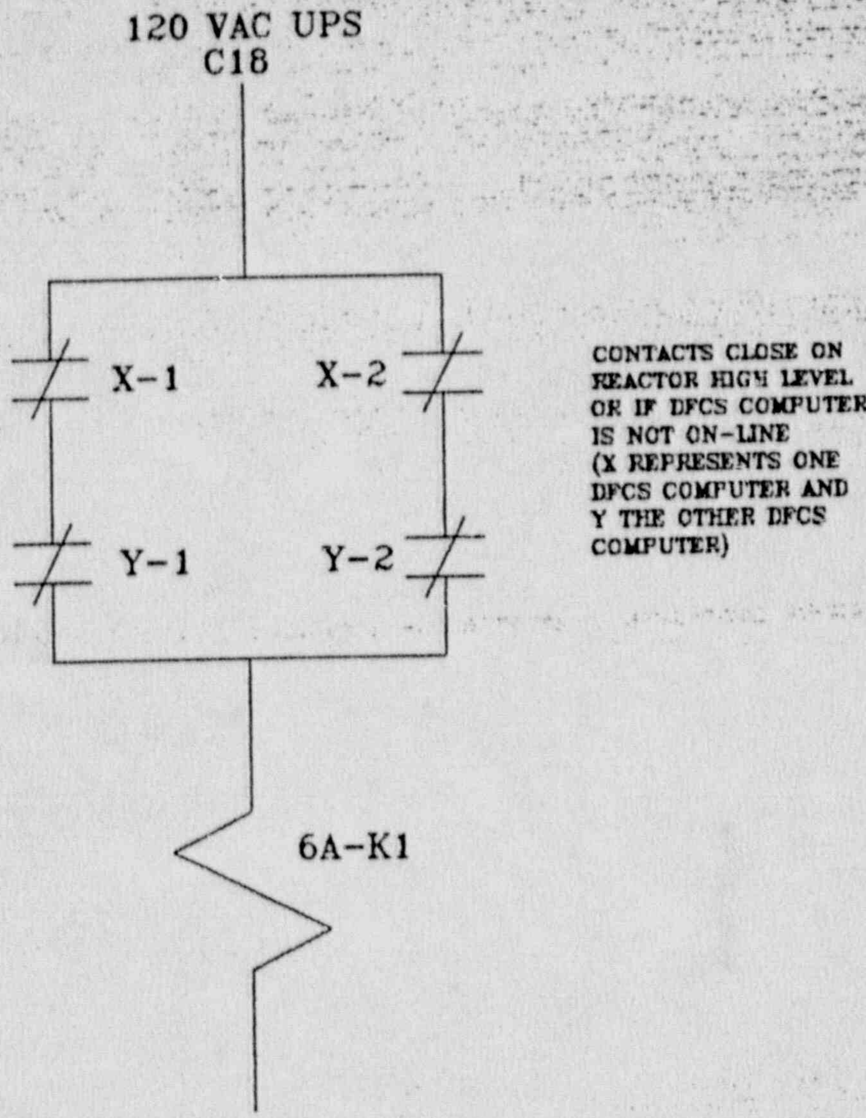

FIGURE 2

MOD 1843 REPLACEMENT
FEEDWATER CONTROL SYSTEM

IMPACT ON RFPT HIGH LEVEL TRIP LOGIC

TJC 1/26/90

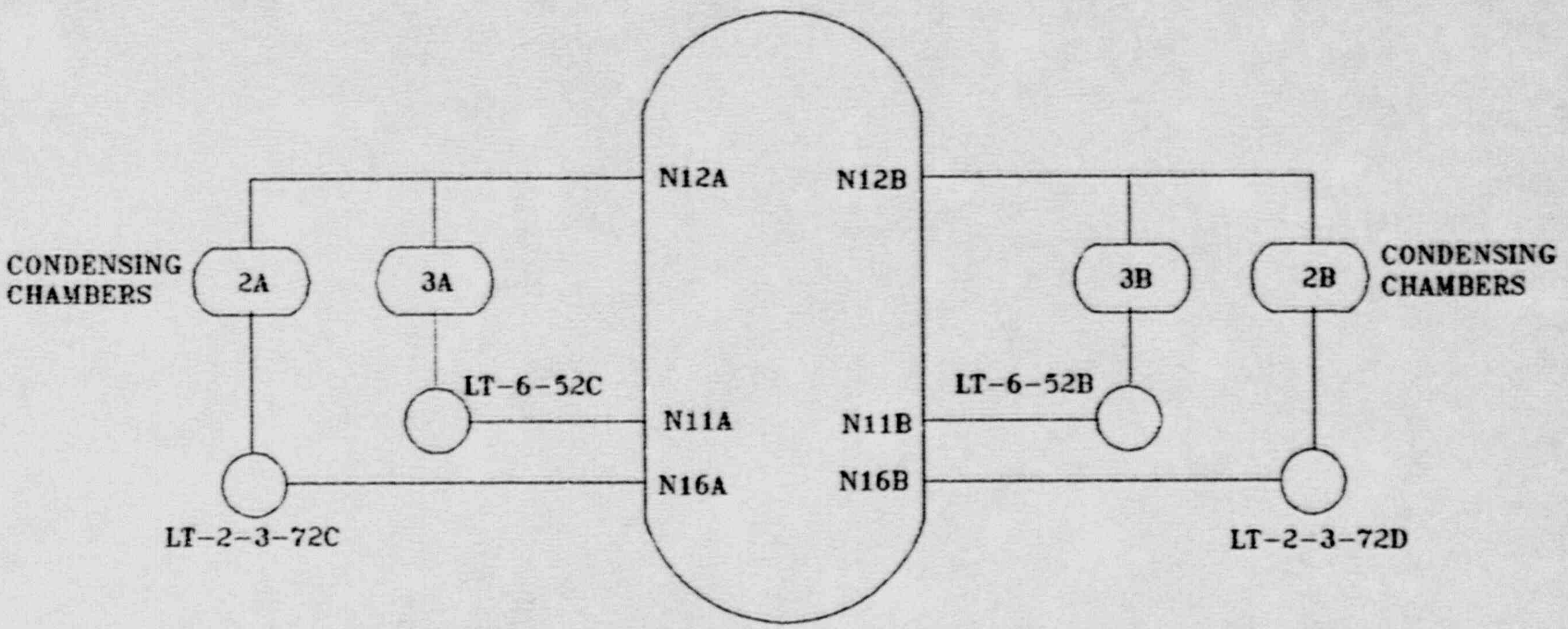

FIGURE #3
ARRANGEMENT OF LEVEL SENSORS
ASSOCIATED WITH THE RFPT HIGH LEVEL TRIP LOGIC