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SHIELDS L. DALTROFF
VICE PRESIDENT
ELECTRIC PRODUCTION

May 13, 1983

Docket Nos. 50-277
50-278

Insp. Rep. 50-277/82-23
50-278/82-22

Mr. Richard W. Starostecki, Director
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Starostecki:

This letter responds to an unresolved item (277/82-23-01, 278/82-22-01) identified in inspection report 50-277/82-23 and 50-278/82-22. The unresolved item deals with four concerns applicable to the design review of plant shielding performed to meet NUREG-0737, Item II.B.2 requirements. These concerns are:

1. Lack of documentation of specifications for the vital areas for PBAPS and the comparison of these areas with the potentially vital areas discussed in NUREG-0737, Item II.B.2.

2. Lack of documentation of projected doses to individuals for necessary occupancy times in vital areas.

3. Lack of documentation on the determination that access is not required to the reactor vessel level instrumentation racks to backfill the instrument line for the reference leg of the instrumentation.

4. Lack of documentation to support the deferral of the modification regarding controls and instrumentation associated with the make-up water supply to the spent fuel pools to permit maintenance of water level from outside secondary containment.

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Attachments 1, 2, 3 and 4 address each of the above concerns, and we believe, provides the information requested by the NRC inspectors. Attachment 5 provides additional information applicable to the first concern. The attached information identifies those areas at Peach Bottom for which accessibility is vital to accident response activities, and provides projected doses to individuals performing necessary functions in these areas. With the implementation of one modification, the dose projections for all areas meet the criteria of NUREG 0737, Item II.B.2. This modification will address the high projected doses to health physics personnel working at the Health Physics Operational Support Center (HP-OSC) during the postulated accident. We plan to provide a backup HP-OSC, or install additional shielding to protect the current HP-OSC, in time for the 1984 emergency drill.

The projected doses include the contribution from airborne activity based on NUREG-0737, Item III.D.3.4, Control Room habitability requirements for all areas except the refueling floor. The airborne dose for the refueling floor is based on Peach Bottom FSAR Design Bases LOCA doses and therefore provides a more realistic estimate of the potential dose than that based on the NRC airborne dose criteria. Both airborne methodologies are more conservative than the NRC's clarification of NUREG-0737, Item II.B.2 source term design criteria for the plant shielding studies provided at the September 27, 1980 Regional Meeting.

Should you have any questions regarding this matter, please do not hesitate to contact us.

Sincerely,

Attachments

cc: R. A. Blough, Site Inspector
Peach Bottom

ATTACHMENT 1

PEACH BOTTOM ATOMIC POWER STATION
RESPONSE TO
UNRESOLVED ITEM (277/82-23-01; 278/82-22-01)

Identification Of Vital Areas And Determination Of Appropriate Types Of Corrective Methods Needed To Provide Access To Vital Areas

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(A) Purpose

(1) To identify Peach Bottom's vital areas, (2) to compare these areas with those specified in NUREG-0737, Item II.B.2, (3) to evaluate formally differences between the two and (4) determine any appropriate types of corrective action needed to provide adequate access to the vital areas.

(B) NRC Finding

REFERENCE: Inspection Report 50-277/82-23; 50-278/82-22

"The licensee's evaluation of plant shielding, as described in BLP-22066 dated May 18, 1982 (Document No. 010878), provided extensive information with respect to dose rate calculations in many plant areas and general information regarding systems assumed to contain high levels of radioactivity in a post-accident situation. However, neither this document nor other licensee documentation provided specification of areas where access is considered necessary for vital system operation after an accident, or an evaluation of all potentially vital areas discussed in NUREG-0737, Item II.B.2. In addition, the licensee did not have documentation that described the projected doses to individuals for necessary occupancy times in vital areas. During discussions with site and corporate office engineering staff personnel and with licensee management, the inspector determined that vital areas were identified and dose levels were calculated during 1979 in response to NUREG-0578, Item 2.1.6.b. This information may have been informally documented. This is supported by the fact that some vital areas were identified and some personnel dose estimates were discussed in various licensee submittals. However, this information apparently was not substantiated by appropriate licensee documentation as discussed below."

With respect to vital area identification, the clarification of NUREG-0737, Item II.B.2 states:

> "The Control Room, Technical Support Center (TSC), sampling station and sample analysis area must be included among those areas where access is considered vital after an accident... The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste panels..."

The referenced NRC inspection report also stated:

> "As stated previously, some of the above areas and other areas were conveyed as vital areas, based on post-accident operational requirements, in various licensee submittals. However, several of the areas identified for consideration in NUREG-0737 Item II.B.2 have not been evaluated formally by the licensee. Therefore, the licensee's shielding design review is incomplete regarding the identification of vital areas and determination of appropriate types of corrective actions needed to provide for adequate access to vital areas. This item is considered unresolved pending completion of licensee actions. (277/82-23-01/ 278/82-22-01)."

(C) Response:

(1) - Peach Bottom Vital Areas

(2) - Comparison of Peach Bottom Vital Areas and Those Identified in NUREG-0737, Item II.B.2

(3) - Evaluated Differences Between (1) and (2)

NUREG 0737 Item II.B.2 Vital Areas	Required Occupancy (Note e)	Applicability To Peach Bottom	Location
Control Room	C	Yes -	El-165'
Technical Support Center	C	Yes -	Unit 1, 3rd Floor

NUREG 0737 Item II.B.2 Vital Areas	Required Occupancy	Applicability To Peach Bottom		Location
Sampling Stations	I	Yes	-	M-G Set Room, El-135' Post LOCA Sampling Station
	I	Yes	-	Rad Effluent Stack Monitor, El-234'
Sampling Analysis	C	Yes	-	Backup Counting Room Unit 1, 1st Floor
	I	Yes	-	Chem Lab, Counting Room, El-116'
Post LOCA Hydrogen Control System	I	Yes	-	CAD Nitrogen Supply Bldg. (Outside Reactor Bldg.) (a)
Containment Isolation Reset Control Area	-	No	-	Not Applicable (b)
Manual ECCS Alignment Area (if any)	-	No	-	Not Applicable (b)
Motor Control Centers	-	No	-	Not Applicable (c)
Instrument Panels	-	No	-	Not Applicable (d)
Emergency Power Supplies	I	Yes	-	Diesel Generator Bldg.
Security Center	I	Yes	-	Guard House
Radwaste Panels	I	Yes	-	Radwaste Control Room, El-135' - Radwaste panels

NUREG 0737 Item II.B.2 Vital Areas	Required Occupancy	Applicability To Peach Bottom	Location
Other Areas	I	Yes -	Makeup water to Spent Fuel Pools (El-234') to maintain water level
	I (f)	Yes -	OSC - (El-135')
	I (f)	Yes -	HP OSC - (El-116')
	I	Yes -	Cable Spreading Room, (El-150')
	I	Yes -	EOP (Unit 1, 2nd floor)

Notes

(a) The CAD system is Peach Bottom's system to maintain hydrogen control after a LOCA. The system is external to secondary containment and the only action required to operate the system is to open the manual isolation valves at the nitrogen tanks in the CAD bldg. (outside the reactor bldg.) and the two solenoid valves in each of the lines to the containment and torus.

(b) All operator operations associated with 1) containment isolation reset control area and 2) manual ECCS alignment area are performed from the main control room; thus, these areas are not vital ones for Peach Bottom.

(c) As previously stated in our correspondence to the NRC (S. L. Daltroff to H. Denton, January 2, 1980), we believe the probability that an operator would have to go to an essential motor control center after an accident is very low...Even if a single failure occurs at a motor control center, we have the capability of removing decay heat; thus, entry into secondary containment would not be necessary. Thus, areas having MCCS are not considered vital ones for Peach Bottom.

(d) The new Peach Bottom emergency procedures are symptom rather than event oriented meaning that operator actions directed by the procedures are based on the status of key plant parameters rather than the expected plant response to a

hypothesized event such as a large break LOCA. These procedures are designed to account for multiple system failures and the operator's inability to perform certain actions by providing the operator with several options for controlling these key plant parameters. Although a few of these options require the operator to perform actions outside the control room, the operator's inability to enter an area due to high radiation would simply result in the selection of another option. Thus, all necessary operator action will be performed from the control room unless the access to the area is not restricted due to high radiation levels.

(e) Required Occupancy:

C - Continuous

I - Infrequent

(f) Alternate location is available in the event continuous occupancy is necessary.

(4) Corrective Action

(a) The maximum whole body dose (180 day TID) to personnel at the HP-OSC (elev.-116') based on the shielding study would be 1600 rem. The high dose is a result of the direct shine through the reactor building personnel access lock at elevation 116' due to the secondary containment airborne activity (based on NUREG 0737 criteria). Either a backup location for the HP-OSC will be selected for use in the event the current HP-OSC is inaccessible due to high radiation, or additional shielding will be installed to protect the current HP-OSC.

(b) Emergency Procedures may be changed to state that the individual(s) who enter(s) the CAD building to open the nitrogen supply valves to initiate the CAD system operation 24 hours after a LOCA should wear Scott air packs.

ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
RESPONSE TO
UNRESOLVED ITEM (277/82-23-01; 278/82-22-01)

Projected Doses To Individuals For Necessary Access To And Occupancy Of Vital Areas Following A LOCA Accident

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REFERENCE: Inspection Report 50-277/82-23; 50-278/82-22

(A) Purpose

Provide the projected doses to individuals for necessary access to and occupancy of the vital areas identified for Peach Bottom following a LOCA.

(B) NRC Finding

Shielding Design Review Verification

Page 8, paragraph 3 of the referenced report states, "...the Licensee did not have documentation that described the projected doses to individuals for necessary occupancy times in vital areas."

(C) Response:

The projected doses to individuals for necessary access to and occupancy of the vital areas identified for Peach Bottom following a LOCA are summarized in the following tables:

Doses for accessing and occupying vital areas requiring continuous occupancy:

Table 1

Control Room (El-165')
TSC, EOF & Backup Counting
Room (Unit 1, 3rd, 2nd
and 1st Floors Respectively)

Doses for accessing and occupying vital areas requiring <u>infrequent</u> occupancy:

<u>Table 2</u>

<u>Within Turbine Hall/Radwaste Bldg. with access from Guard House</u>:

Health Physics - Operational Support Center (HP-OSC) (El-116')
Operational Support Center (OSC) (El-135')
Chem Lab/Counting Room (El-135')
Radwaste Control Room (El-135) - Radwaste Panels
M-G Set Room (El-135') - Post LOCA Sampling
Cable Spreading Room (El-150')

<u>Table 3</u>

OSC (El-135') to Diesel Generator Building

<u>Table 4</u>

OSC (El-135') to CAD Nitrogen Supply Building

<u>Table 5</u>

TSC (Unit 1) to Rad Stack Effluent Monitor (El-234') - replace iodine cartridge (NUREG 0737, Item II.F.1)

<u>Table 6</u>

OSC (El-135') to spent fuel pools (El-234') to maintain water level of the spent fuel pools.

<u>Table 7</u>

Post LOCA Sampling (NUREG-0737, Item II.B.3)

Dose rate maps for potentially occupied areas are not provided. Maximum dose rates for occupying and accessing the vital areas were utilized and are presented in the attached tables.

The following summarizes the projected doses (Tables 1B thru 7) to individuals accessing the Vital Areas:

Access Route	See Table	Time After Accident	Projected Total Whole Body Dose Rem
. Guard House to TSC, EOF and Backup Counting Room	1B	8 hrs.*	0.175
. Guard Hose to Control Room (El-165')	1B	8 hrs.*	0.288
. Within Turbine Hull/ Radwaste Building Complex (HP-OSC Chem Lab/ Counting Room, OSC, M-G Set Room, Radwaste Control Room, and Cable Spreading Room)	2	8 hrs.*	0.012
. OSC to Diesel-Generator Building	3	24 hrs.	0.422
. OSC to CAD Building	4	24 hrs.	3.621
. TSC to El-234' - Cartridge Exchange at Rad. Effluent Monitor	5	1 hr.	4.981
. TSC to El-234' - Makeup Water to Spent Fuel Pools	6	2 hrs.	4.952
. Post Accident Sampling Capability NUREG-0737, Item II.B.3 - TSC to M-G Set Room	7	1 hr.	1.174

*Maximum dose rate following the accident

TABLE 1 (A) PROJECTED DOSES (REM) TO INDIVIDUALS FOR NECESSARY OCCUPANCY TIMES IN VITAL AREAS REQUIRING CONTINUOUS OCCUPANCY

Projected TID Doses (180 days)

Vital Areas	Thyroid (Rem)	Total Whole Body (Rem)	Skin (Rem)
Control Room (CR) (El-165')	1.6	0.012	0.1
Backup Counting Room EOF (Unit 1 - 1st Floor)	0.32	3.2	0.049
TSC (Unit 1 - 3rd Floor)	0.32	3.2	0.049
EOF (Unit 1 - 2nd Floor)	0.32	3.2	0.049

Notes:

(1) Continuous Occupancy = 100% 0-1 day, 60% 1-4 days, 40% 5-180 days

(2) These vital areas are in conformance with NUREG 0737, Item III.D.3.4 - Control Room Habitability Requirements

(3) Reference: BLP-22066, dated May 18, 1982

TABLE 1 (B) PROJECTED DOSES TO INDIVIDUALS ACCESSING THE VITAL AREAS REQUIRING CONTINUOUS OCCUPANCY - (ENTRY 24 HOURS AFTER LOCA)

Travel Route	Time (min)	TWB Dose Rates (Rem/Hr)	TWB Dose (Rem)
Shine			
Outside Guard House (G.H.) (1)	1	0.655	0.011
Inside Guard House (G.H.) (2)	2	1.2	0.040
G.H. to Turbine Hall Rolling Door (2)	3	2.63	0.131
Airborne			
Outside Time (6 min) + Access Time to Control Room (10 min)	16	0.092	0.024
TOTAL ONE WAY TRIP	16	--	0.206

(1) Reference BLP-22191, dated February 11, 1983
(2) Reference BLP-22066, dated May 18, 1982
(3) Total whole body dose - one way trip

	(CR)	TSC/EOF (4)
Highest dose within 1st 24 hours	0.288 Rem	0.149 rem
24 hours after LOCA	0.206 Rem	0.109 rem
4 days after LOCA	0.082 Rem	0.040 rem

(4) Includes 10 minutes travel time between TSC/EOF and Guard House

TABLE 2 - PROJECTED DOSES TO INDIVIDUALS FOR NECESSARY ACCESS TO AND INFREQUENT OCCUPANCY OF VITAL AREAS WITHIN TURBINE HALL/RADWASTE BUILDING COMPLEX (EL-116' to EL-165')

The vital areas within the Turbine Hall/Radwaste Building Complex required for necessary access to and infrequent occupancy include the following:

HP-OSC (El-116') Note (1)
Chem Lab/Counting Room (El-135')
OSC (El-135')
M-G Set Room (El-135') - Post LOCA Sampling
Radwaste Control Room (El-135') - Radwaste Panels
Cable Spreading Room (El-150')

The only significant radiation source that individuals would receive in accessing and occupying the above areas required for infrequent occupancy is the airborne component associated with unfiltered areas. The projected Total Whole Body (TWB) and thyroid dose to the individuals in 1) making the most time consuming trip (10 minutes) between the vital areas within the complex (rolling steel door at centerline of Turbine Hall (El-116') and the control room (El-165') and 2) occupying the vital areas for eight (8) hours are tabulated below for selected times after the accident.

Time After LOCA	One Way Travel Time-(10 min) TWB(1) (rem)	One Way Travel Time-(10 min) Thyroid(2) (rem)	Continuous 8 Hr Occupancy TWB(1) (rem)	Continuous 8 Hr Occupancy Thyroid(2) (rem)
x*	0.033	0.012	1.6	0.584
24 hours	0.015	0.012	0.736	0.584
4 day	0.004	0.067	0.192	3.2
10 day	0.002	0.231	0.096	11.0

x*= Highest dose rate in first 24 hours after LOCA

REFERENCE: (1) - BLP-22066, dated May 18, 1983
(2) - BLP-22191, dated February 11, 1983
and Telecon (Bechtel to EJP) March 10, 1983

Note (1) Assumes HP-OSC used is not in room 124 near the personnel access lock to the reactor building. The maximum dose of the current HP-OSC location would be 1600 rem.

TABLE 3 - PROJECTED TOTAL WHOLE BODY DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF VITAL AREA - DIESEL-GENERATOR BUILDING, OUTSIDE TURBINE HALL (ENTRY 24 HOURS AFTER LOCA)

Travel Route	Duration (min)	Dose Rate TWB (Rem/Hr)	TWB Dose (Rem)
Shine			
OSC (El-135') to Turbine Hall Rolling Steel Door	5	--	--
Door Past Adm. Bldg.	1.5	2.63	0.0658
Adm. Bldg. to Steps of Diesel-Gen. Bldg.	1.5	4.37	0.1093
One Way Trip	8	--	0.175
Round Trip	16		0.350
Inside Diesel-Gen. Bldg	30	0.004	0.002
TOTAL SHINE (Round Trip)	46	--	0.352
Airborne	46	0.092	0.070
TOTAL TWB DOSE	46	--	0.422

Projected doses to individuals who may access and occupy the Diesel-Generator Building after the first 24 hours will have lower projected doses as follows:

Time After LOCA	TWB Dose (rem)
4 days	0.154
10 days	0.073

REFERENCE: BLP-22066, dated May 18, 1982
BLP-22191, dated February 11, 1983

TABLE 4 - PROJECTED TOTAL WHOLE BODY DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF VITAL AREA - CAD BUILDING, OUTSIDE REACTOR BUILDING (ENTRY 24 HOURS AFTER LOCA)

Travel Route	Duration (min)	Dose Rate TWB (Rem/Hr)	TWB Dose (Rem)
Shine			
OSC (EL-135') to Turbine Hall Rolling Steel Door	5	--	--
Door Past Adm. Bldg.	1.5	2.63	0.066
Adm. Bldg to Diesel-Gen. Bldg. Steps	1.5	4.37	0.109
Steps to Corner of Aux. Blrs.	1.5	11.6	0.290
Corner Aux. Blrs to CAD Bldg.	1.5	18.8	0.470
One Way Trip	11	--	0.935
Round Trip	22	--	1.870
Outside CAD Bldg.	20 sec	129.	0.717
Inside CAD Bldg.	5	7.61	0.634
Outside CAD Bldg.	10 sec	129.	0.358
TOTAL SHINE (Round Trip)	27.5	--	3.579
Airborne	27.5	0.092	0.042
TOTAL TWB DOSE (Rem) (Round Trip)	27.5	--	3.621

REFERENCE: BLP-22061, dated May 18, 1982
BLP-22191, dated February 11, 1983

TABLE 5 - PROJECTED DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF 'VITAL' AREAS - CARTRIDGE EXCHANGE AT RAD. EFFLUENT STACK MONITOR (EL-234') (ENTRY ONE HOUR AFTER LOCA)

Radiation Component	Time of Entry (min)	Total Whole Body Dose Rate (Rem/Hr)	Time Duration (min)	Dose (Rem)	Total Dose (Rem)
Shine (TSC to Turbine Hall)					
TSC to Guard House	60	0.494	10	0.082	
Guard House		0.870	2	0.029	
Guard House to Turbine Hall		1.78	3	0.089	
SUB TOTAL			15	-----	0.200
Airborne (TSC to El-165')	60	0.130	25*	-----	0.054
Shine (El 165' - El 234')					
El-165' to El-195'	60	12.0	5	1.000	
El-195' to El-234'	65	1.0	5	0.083	
El-234	70	0.038	10	0.006	
El-234' to El-195'	80	0.95	5	0.079	
El-195' to El-165'	85	9.5	5	0.790	
SUB TOTAL			30	-----	1.958
Containment Atmosphere (FSAR)					
Immersion (N.G.)	60	2.09 rad/hr	30	1.045	
(I-131)		.042 rad/hr	30	0.021	
SUB TOTAL			30	-----	1.066
Shine (Turbine Hall to TSC)					
Turbine Hall To Guard House	90	2.13	3	0.107	
Guard House		1.04	2	0.035	
Guard House to TSC		0.588	10	0.098	
SUB TOTAL			15	-----	0.240
Airborne (El-165' to TSC)	90	0.15	25*	-----	0.063
TOTAL	--		80		3.581
Cartridge Exchange & Transport					1.400
GRAND TOTAL (Round Trip) (Rem)			80		4.981

*Includes 10 minutes of travel time between Turbine Hall door and El-165'

REFERENCE: BLP-22061 (5/18/82) and BLP-22191 (2/11/83)
FSAR Table 14.6.5

TABLE 6 - PROJECTED DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF 'VITAL' AREAS - MAKEUP WATER TO SPENT FUEL POOLS (EL-234') TO MAINTAIN WATER LEVEL FOLLOWING A LOCA (ENTRY TWO HOURS AFTER LOCA)

	Total Whole Body				
Radiation Component	Time of Entry (min)	Dose Rate (Rem/HR)	Time Duration (min)	Dose (Rem)	Total Dose (Rem)
Shine (TSC to Turbine Hall)					
TSC to Guard House	120	0.681	10	0.114	
Guard House		1.2	2	0.040	
Guard House to Turbine Hall		2.47	3	0.124	
SUB TOTAL		-----	15	-----	.278
Airborne (TSC to El 165')		0.17	25*	-----	.071
Shine (El 165' - El 234')					
El-165' to El-195'	120	8.1	10	1.35	
El-195' to El-234'		0.76	10	0.13	
El-234'		-----	30	-----	
El-234' to El-195'		0.62	5	0.05	
El-195' to El-165'		6.59	5	0.55	
SUB TOTAL		-----	60	-----	2.080
Containment Atmosphere (FSAR)					
Immersion (N.G.)	120	2.09	60	2.090	
(I-131)		0.042		0.042	
SUB TOTAL		-----	60	-----	2.132
Shine (Turbine Hall to TSC)					
Turbine Hall To Guard House	180	2.81	3	0.141	
Guard House		1.35	2	0.045	
Guard House to TSC		0.767	10	0.128	
SUB TOTAL	---	-----	15	-----	.314
Airborne (El-165' to TSC)	180	0.185	25*	-----	.077
GRAND TOTAL (Round Trip)	---	-----	110	-----	4.952

*Includes 10 minutes travel time between Turbine Hall Door and El-165'

REFERENCE: BLP-22061 (5/18/82) and BLP-22191 (2/11/83)

TABLE 7 - Post-Accident Sampling Capability NUREG-0737, Item II.B.3 (To Verify Compliance with GDC 19 For A Sample Taken 1 Hour After An Accident)

Whole Body Dose Assessment

	Time (min)	Background Dose (Rem/Hr)	Sample Dose (Rem/Hr)	Integrated Dose (Rem)
Liquid Sample				
a) recirculate sample	10	0.568	0.066	.106
b) operate station	5	0.568	0.100	.056
c) transport sample cask	20	0.883	0.006	.296
d) handle sample	10 sec.	0.059	0.161	.001
e) analyze sample	20	0.059	0.080	.046
			Total	.505
Gas Sample				
a) recirc. sample	20	0.568	0.002	.190
b) operate station	5	0.568	0.360	.077
c) handle bottle	1	0.568	0.410	.016
d) transport sample cask	20	0.883	0.054	.312
e) analyze sample	20	0.059	0.020	.026
			Total	.0621
Particulate/Iodine				
a) recirculate sample	20	0.568	0.002	.190
b) operate station	5	0.568	0.360	.077
c) transport cartridge	20	0.883	0.890	.591
d) analyze sample	20	0.059	0.890	.316
			Total	1.174

Some additional dose to extremeties will result from the limited handling of samples in the laboratory. Because of the use of sample dilutions, small volume samples, shielded casks, lead brick piles, and laboratory extension devices (i.e. - tongs) doses to the extremeties are estimated to be 100 to 200 MR for each sample.

REFERENCE: S. L. Daltroff (PECO) to J. F. Stolz (NRC) Letter Dated January 31, 1983.

ATTACHMENT 3

PEACH BOTTOM ATOMIC POWER STATION
RESPONSE TO
UNRESOLVED ITEM (277/82-23-01; 278/82-22-01)

Backfilling Reference Legs Of Reactor
Water Level Instrumentation

(A) Purpose

To document the record on why it is not necessary to backfill the reference legs to the reactor water level instrumentation following a LOCA

(B) Background

S. L. Daltroff letter to H. Denton, dated January 2, 1980 (Subject: Design Review Studies Required by Short Term Lessons Learned)

"General Electric Company is evaluating the effects of an accident on the reactor vessel level instrumentation as part of NUREG-0578, Item 2.1.3.b. Included in this eveluation is the determination of whether access is required to the reactor vessel level instrument racks to backfill the instrument line for the reference leg of the instrumentation. This evaluation is expected to be complete at the end of the year. If access is required, we will provide a means of backfilling which is operable from an accessible area. This modification, if necessary, will be completed prior to January 1, 1981."

NRC Finding

Ref: NRC Inspection Report 50-277/82-23; 50-278/82-22

d. Vital Area Accessibility - Procedure Review

"The inspector reviewed two emergency procedures that would be implemented by the licensee in the

event of various severities of loss of coolant accidents. The review included (1) a plant walkdown of portions of each procedure to determine the ability to perform the procedure and the accessibility of manual valves that may require local operation, and (2) an assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included Emergency Procedure E-14 "Large Break-Loss of Coolant Accedent - Offsite Power Available," Revision 15 dated May 17, 1982, and Emergency Procedure E-15 "Loss of Coolant Accedent Concurrent with Loss of Offsite Power - Loss of All Seismic Class II Equipment - Failure of One Diesel Generator to Start," Revision 14 dated May 19, 1982.

Follow Action step 14 of Emergency Procedure E-14 states: "Notify (I&C) Lab to backfill (reactor) level instrumentation lines. This will provide reliable reactor vessel level instrumentation." The inspector noted that this action would be performed at the 165' elevation of the Reactor Building, which may be inaccessible due to post-accident high radiation conditions. However, procedural controls have not been established to provide the methods (pre-planned access route, instructions for valve operations, etc.) for backfilling the instrument lines. The licensee's submittal to the NRC dated January 2, 1980, stated that General Electric Company was evaluating the effects of an accident on reactor vessel instrumentation, including the determination of whether access is required to the reactor vessel level instrument racks to backfill the instrument reference legs. If access was required, a means of backfilling was to be provided by January 1, 1981. The licensee evaluation of this matter and determination of corrective actions has not been completed."

(C) <u>Response</u>

In NEDE 24801 entitled "Review of BWR Reactor Vessel Water Level Measurement" (Proprietary), dated April, 1980, General Electric Company states (page 2.27) that "filling the vessel with relatively low-temperature water", (from the low-

pressure ECC Systems), "would refill the reference and variable legs of the instruments, thus restoring the capability of accurate measurement if the levels were subsequently reduced to within instrument range...", and concludes, "Therefore ... the error introduced by drywell heatup and in reactor pressure is of significance only if reactor operations do not flood the steam lines with the low-pressure system following the accident."

With respect to maintaining core cooling following a LOCA, Emergency Procedure E-14 provided the following:

1- states that one of the objectives in the event of a pipe break is to maintain core cooling (page 2)

2- requires the operator to add water to the reactor vessel (page 3, Table A description)

3- states that filling the vessel completely is desirable (page 6, below Caution #9)

4- warns the operator not to trust the level indication if the system has been depressurized quickly or if the level indication looks 'eratic' (page 5, Caution #7)

5. instructs operator to continuously monitor vessel level and pressure from multiple indications (page 3, Caution #2)

6. states that ECCS system cannot be shutdown unless there are multiple confirming process parameter indications (such as level indications from several instruments) that the core and containment are in safe, stable condition (page 4, Caution #5).

Emergency Procedure E-14 reflected PECo's position that reactor flooding may continue indefinitely until such time that 1) the process parameter indications (such as level indications from several instruments) confirm that the core and containment are in a safe, stable condition, or 2) that the areas containing the reference legs to be backfilled are accessible. Back filling is a conservative measure to assure reliable level indication following conditions which could have resulted in reference leg flashing; it is not required to ensure safe plant operation. This action can be used as part of the confirming process that ensures the core and

containment are in safe, stable conditions. Procedure E-14 has recently been superseded by symptomatic procedures to meet the requirements of NUREG-0737 Supplement 1.

The new Peach Bottom emergency procedures are symptom rather than event oriented meaning that operator actions directed by the procedures are based on the status of key plant paramaters rather than the expected plant response to a hypothesized event such as a large break LOCA. These procedures are designed to account for multiple system failures and the operator's inability to perform certain actions by providing the operator with several options for controlling these key plant parameters. Although a few of these options require the operator to perform actions outside the control room, the operator's inability to enter an area due to high radiation would simply result in the selection of another option. In order to demonstrate this flexibility and to show why this area is not vital, design basis accident scenarios have been applied to these procedures. Using these procedures, the operator would consider performing the following action outside the control room:

ACTION:	backfilling level instrument reference legs
PURPOSE:	restore reliable level indication before reactor flooding is terminated.
REFERENCE:	T-116 Rev. G step RF-16
AREAS OUTSIDE CONTROL ROOM:	R.B. Elev. 165' and 135'
ANALYSIS:	A. Backfilling reference legs is only required prior to termination of reactor flooding if other means cannot be used to confirm that the core and containment are in a safe, stable condition. Reactor flooding may continue indefinitely until these areas are accessible.
	B. Backfilling is a conservative measure to assure reliable level indication following conditions which could have resulted in reference leg flashing. It is probable that these legs will refill

in the process of flooding the reactor.

Thus, access is not required to the reactor vessel level instrument racks to backfill the instrument reference legs following a LOCA.

ATTACHMENT 4

PEACH BOTTOM ATOMIC POWER STATION
IN RESPONSE TO
UNRESOLVED ITEM (277/82-23-01; 278/82-22-01)

Documentation Of The Deferral Of Modification to Controls And Instrumentation To Makeup Water Supply To Spent Fuel Pools (E1-234')

(A) Purpose

To document the deferral of the modification of the control and instrumentation associated with the makeup water supply to the spent fuel pools (E1-234') to permit maintenance of pool water level from outside secondary containment following a LOCA accident.

(B) Background

REFERENCE: Inspection Report 50-277/82-23;50-278/82-22

(1) "S. L. Daltroff letter to H. Denton, dated January 31, 1980 (Subject: Design Review Studies Required by Short Term Lessons Learned):

With respect to areas requiring infrequent access, revised calculations indicated that secondary containment would be inaccessible for several days, and two modifications must be made regarding 1) the capability to obtain post-accident primary coolant and primary containment samples, and 2) controls and instrumentation associated with the makeup water supply to the spent fuel pools to permit maintenance of pool water level from outside secondary containment. Both modifications were to be completed by January 1, 1981, unless precluded by equipment unavailability."

(2) "S. L. Daltroff letter to D. Eisenhut, dated October 15, 1980 (Subject: Implementation Of NRC Action Plan Requirements):

With respect to the plant shielding study, the licensee discussed the relocation of facilities and equipment, proposed for completion by January 1, 1981 (as presented in the January 31, 1980 submittal) and specifically noted this involves relocation of the spent fuel makeup controls to areas outside the Reactor Building. The licensee

stated further that an NRC Region I meeting held in Arlington, Virginia on September 22, 1980 provided additional clarification of the source term design criteria for the plant shielding study. The licensee's reassessment of the shielding study, based on this new clarification, indicated that post-accident radiation conditions will not impact on reactor building accessibility. Therefore, the licensee proposed that implementaion of the modifications described above be deferred until such time that their need is clearly established."

(3) "S. L. Daltroff letter to D. Eisenhut, dated January 8, 1981 (Subject: Information Requested by NUREG-0737) states in part:

Although NUREG-0737 Item II.B.2 required no responses from licensees of operating reactors (unless deviations to the position or clarification were necessary), this submittal stated, in part:"

"Based upon the clarified source term design criteria and the expanded vital area criteria of NUREG-0737, the results presented in our submittal of January 31, 1980, S. L. Daltroff to H. R. Denton, indicate that the post-accident radiation conditions will not impact on accessibility to vital areas defined for PBAPS (Peach Bottom Atomic Power Station)."

(4) NRC Finding

"As described in paragraph 3.b.(2), the licensee committed (January 31, 1980 submittal) to completing a modification regarding the controls and instrumentation associated with the make-up water supply to the spent fuel pools to permit maintenance of water level from outside secondary containment. As noted in paragraph 3.b.(3), the licensee proposed (October 15, 1980) that implementation of this modification be deferred until such time as the need is clearly established. The basis for deferral, as stated by the licensee, was that reassessment of the shielding study, based on additional clarification of the source term design criteria provided during a September 22, 1980 meeting with the NRC, indicated that post-accident radiation conditions will not impact on

reactor building accessibility. The "additional clarification" which led to the licensee's conclusion was not described further in the licensee's submittal. The licensee's shielding design review discussed in paragraph 3.c. does not support this conclusion, in that data for several areas of the reactor building indicate very high post-accident dose rates due to equipment/piping shine. Based on the inspector's review of the shielding design review data, the licensee's general statements (January 8, 1981 submittal) that "post-accident radiation conditions will not impact on accessibility to vital areas defined for PBAPS," as discussed in paragraph 3.b.(4), and (April 15, 1982 submittal) that the "current design provides access to vital areas under accident conditions," as described in paragraph 3.b.(5), also appear to be unsupported. The licensee's specific evaluation of completing a modification to permit post-accident maintenance of spent fuel pool water level from outside secondary containment and the general evaluation that current design provides access to vital areas is considered part of the unresolved item discussed in paragraph 3.c (277/82-23-01; 278/82-22-01)."

(C) Response

(1) Additional Clarification Of NRC Region I Meeting Held In Arlington, Virginia On September 22, 1980

REFERENCE: E. C. Kistner to J. S. Kemper letter dated October 16, 1980

The additional clarification provided at the Arlington, Virginia meeting was that in the assessment of doses to the individual for necessary occupancy of vital areas under II.B.2, airborne doses were to be ignored. Thus with respect to access to the refueling floor (El-234') in the reactor building if the cloud in secondary containment is not considered, then access to the spent fuel pools (to maintain pool water level) is not precluded. Similarly, access to the Rad effluent stack monitor is also not precluded.

(2) Analysis - II.B.2 Plant Shielding Plus FSAR Airborne

In response to a telephone conversation between PECo (A. J. Marie and W. Birely) and the NRC, in July of 1981 with regard to the access to the Rad effluent stack monitor on El-234' of the refueling floor (NUREG-0737 Item II.F.1), PECo provided a projected dose to the individual who replaces an iodine cartridge 1 hour after a LOCA based upon the analysis that utilized the II.B.2 shine and the FSAR Airborne dose (Table 5 of Attachment 2).

Using the same methodology to access to the spent fuel pools (El-234') 2 hours after a LOCA, the individual is projected to receive a dose of less than 5 rem (Table 6 of Attachment 2).

TABLE 5 - PROJECTED DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF 'VITAL' AREAS - CARTRIDGE EXCHANGE AT RAD. EFFLUENT STACK MONITOR (EL-234') (ENTRY ONE HOUR AFTER LOCA)

	Total Whole Body				
Radiation Component	Time of Entry (min)	Dose Rate (Rem/Hr)	Time Duration (min)	Dose (Rem)	Total Dose (Rem)
Shine (TSC to Turbine Hall)					
TSC to Guard House	60	0.494	10	0.082	
Guard House		0.870	2	0.029	
Guard House to Turbine Hall		1.78	3	0.089	
SUB TOTAL			15	-----	0.200
Airborne (TSC to El-165')	60	0.130	25*	-----	0.054
Shine (El 165' - El 234')					
El-165' to El-195'	60	12.0	5	1.000	
El-195' to El-234'	65	1.0	5	0.083	
El-234	70	0.038	10	0.006	
El-234' to El-195'	80	0.95	5	0.079	
El-195' to El-165'	85	9.5	5	0.790	
SUB TOTAL			30	-----	1.958
Containment Atmosphere (FSAR)					
Immersion (N.G.)	60	2.09 rad/hr	30	1.045	
(I-131)		.042 rad/hr	30	0.021	
SUB TOTAL			30	-----	1.066
Shine (Turbine Hall to TSC)					
Turbine Hall To Guard House	90	2.13	3	0.107	
Guard House		1.04	2	0.035	
Guard House to TSC		0.588	10	0.098	
SUB TOTAL			15	-----	0.240
Airborne (El-165' to TSC)	90	0.15	25*	-----	0.063
TOTAL	--		80		3.581
Cartridge Exchange & Transport					1.400
GRAND TOTAL (Round Trip) (Rem)			80		4.981

*Includes 10 minutes of travel time between Turbine Hall door and El-165'

REFERENCE: BLP-22061 (5/18/82) and BLP-22191 (2/11/83)
FSAR Table 14.6.5

TABLE 6 - PROJECTED DOSE TO INDIVIDUALS FOR NECESSARY ACCESS TO AND OCCUPANCY OF 'VITAL' AREAS - MAKEUP WATER TO SPENT FUEL POOLS (EL-234') TO MAINTAIN WATER LEVEL FOLLOWING A LOCA (ENTRY TWO HOURS AFTER LOCA)

	Total Whole Body				
Radiation Component	Time of Entry (min)	Dose Rate (Rem/HR)	Time Duration (min)	Dose (Rem)	Total Dose (Rem)
Shine (TSC to Turbine Hall)					
TSC to Guard House	120	0.681	10	0.114	
Guard House		1.2	2	0.040	
Guard House to Turbine Hall		2.47	3	0.124	
SUB TOTAL		-----	15	-----	.278
Airborne (TSC to El 165')		0.17	25*	-----	.071
Shine (El 165' - El 234')					
El-165' to El-195'	120	8.1	10	1.35	
El-195' to El-234'		0.76	10	0.13	
El-234'		-----	30	-----	
El-234' to El-195'		0.62	5	0.05	
El-195' to El-165'		6.59	5	0.55	
SUB TOTAL		-----	60	-----	2.080
Containment Atmosphere (FSAR)					
Immersion (N.G.)	120	2.09	60	2.090	
(I-131)		0.042		0.042	
SUB TOTAL		-----	60	-----	2.132
Shine (Turbine Hall to TSC)					
Turbine Hall To Guard House	180	2.81	3	0.141	
Guard House		1.35	2	0.045	
Guard House to TSC		0.767	10	0.128	
SUB TOTAL	---	-----	15	-----	.314
Airborne (El-165' to TSC)	180	0.185	25*	-----	.077
GRAND TOTAL (Round Trip)	---	-----	110	-----	4.952

*Includes 10 minutes travel time between Turbine Hall Door and El-165'

REFERENCE: BLP-22061 (5/18/82) and BLP-22191 (2/11/83)

ATTACHMENT 5

PEACH BOTTOM ATOMIC POWER STATION
IN RESPONSE TO
UNRESOLVED ITEM (277/82-23-01; 278/82-22-01)

Assessment of Areas Identified in Emergency Procedures As Not 'Vital' Ones Pursuant to NUREG-0737, Item II.B.2

(A) Purpose

To document the record on why other areas (besides backfilling reference legs of reactor water level instrumentation) identified in the Emergency Procedure are not 'vital' ones pursuant to NUREG-0737, Item II.B.2.

(B) NRC Finding

Reference: Inspection Report 50-277/82-23; 50-278/82-22

"d. Vital Area Accessibility - Procedure Review

The inspector reviewed two emergency procedures that would be implemented by the licensee in the event of various severities of loss of coolant accidents. The review included: (1) a plant walkdown of portions of each procedure to determine the ability to perform the procedure and the accessibility of manual valves that may require local operation, and (2) the assessment of potential exposures to plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included Emergency Procedure E-14 "Large Break-Loss of Coolant Accident - Offsite Power Available," Revision 15 dated May 17, 1982, and Emergency Procedure E-15 "Loss of Coolant Accident Concurrent with Loss of Offsite Power Loss of All Seismic Class II Equipment - Failure of One Diesel Generator to Start," Revision 14 dated May 19, 1982.

Followup Action step 14 of Emergency Procedure E-14 states: "Notify (I&C) Lab to backfill (reactor) level instrumentation lines. This will provide reliable reactor vessel level instrumentation." The inspector noted that this action would be performed at the 165' elevation of the Reactor Building, which may be inaccessible due to post-accident high radiation conditions. However, procedural controls have not been established to provide the method (pre-planned access route, instructions for valve operations, etc.) for backfilling the instrument lines. The licensee's

submittal to the NRC dated January 2, 1980, stated that General Electric Company was evaluating the effects of an accident on reactor vessel instrumentation, including the determination of whether access is required to the reactor vessel level instrument racks to backfill the instrument reference legs. If access was required, a means of backfilling was to be provided by January 1, 1981. The licensee evaluation of this matter and determination of corrective actions has not been completed. This matter is discussed further in paragraph 3.f.(1)."

"f. Findings

(1) As described in paragraph 3.b.(1), the licensee committed (January 2, 1980 submittal) to evaluating the need for access to backfill reactor vessel instrument lines and, if necessary, to provide a means for backfilling from an accessible area. As noted in paragraph 3.b(5), the licensee subsequently concluded (April 15, 1982 submittal) that the current design provides adequate access to vital areas, however, the evaluation of backfilling instrument lines was not specifically discussed. As discussed in paragraph 3.d., Emergency Procedure E-14 specifies backfilling the instrument lines as a followup action for a loss of coolant accident, however,no provisons have been included for performing this operation from an accessible area. Licensee Evaluation of backfilling the instrument lines and determination of appropriate corrective actions (design change, increased permanent or temporary shielding, or post-accident procedural controls) is considered part of the unresolved item discussed in paragraph 3.c. (277/82-23-01); 278/82-22-01).

C. Response

Emergency procedure E-14 and E-15, referenced in the inspection report, have been superseded by new symptomatic emergency procedures as required by NUREG-0737, Supplement 1. The new emergency procedures are symptom rather than event oriented meaning that operator actions directed by the procedures are based on the status of key plant parameters rather than the expected plant response to a hypothesized event such as a large break LOCA. These procedures are designed to account for multiple system failures and the operator's inability to perform certain actions by providing the

operator with several options for controlling these key plant parameters. Although a few of these options require the operator to perform actions outside the control room, the operator's inability to enter an area due to high radiation would simply result in the selection of another option. The following assessment provides justification for not identifying areas associated with the emergency procedures as vital areas pursuant to NUREG-0737, Item II.B.2.

1. ACTION: bypassing the drywell cooler fan trips
   PURPOSE: control drywell temperature
   REFERENCE: T-102 Rev. G step DW/T-5
   AREAS OUTSIDE CONTROL ROOM:

Unit 2		Unit 3	
Radwaste	El. 116'	Radwaste	El 116'
R.B.	El. 135'	R.B.	El.135'
R.B.	El. 165'	Cable Spreading Room	

   ANALYSIS:
   A. Using the drywell cooler fans is one of several options for controlling drywell temperature. The others include the use of drywell sprays and reactor depressuziation.

   B. FSAR analysis of the large break LOCA assumes that these fans trip and are not restored to service, therefore they are not essential to the mitigation of that accident.

2. ACTION: backfilling level instrument reference legs
   PURPOSE: restore reliable level indication before reactor flooding is terminated.

REFERENCE: T-116 Rev. G step RF-16
AREAS OUTSIDE CONTROL ROOM: R.B. Elev. 165' and 135'
ANALYSIS:
A. Backfilling reference legs is only required prior to termination of reactor flooding if other means cannot be used to confirm that the core and containment are in a safe stable condition. Reactor flooding may continue indefinitely until these areas are accessible.

B. Backfilling is a conservative measure to assure reliable level indication following conditions which could have resulted in reference leg flashing. It is probable that these legs will refill in the process of flooding the reactor.

3. ACTION: obtaining a torus water sample prior to discharging torus water; Control torus water level

   REFERENCE:T-102 Rev. G section T/L
   AREA OUTSIDE THE CONTROL ROOM: T.S. Elev. 135'
   ANALYSIS: A. This sample can be obtained in an area which is currently designated as vital.
   B. Increasing torus level can be controlled by terminating makeup sources external to primary containment.
   C. The design basis accidents do not result in unacceptably high torus levels.

4. ACTION: adjusting the Standby Gas Treatment System (SBGTS) dampers
   PURPOSE: control containment depressurization
   REFERENCE: T-102 Rev. G step DW/P-4
   AREA OUTSIDE THE CONTROL ROOM: B.W. El. 91'6"
   ANALYSIS: A. using SBGTS to control drywell pressure is one of several options for controlling drywell pressure. The others include reactor depressurization, containment sprays, reactor flooding and primary containment venting.
   B. the inability to adjust these dampers does not preclude batch venting through standby gas.
   C. the damper control is automatically bypassed and the damper position is fixed by the PCIS logic during a design basis LOCA.

The new emergency procedures have also been designed to specify actions for the complete spectrum of emergencies including emergencies beyond the design basis. These procedures specify appropriate actions for any mechanically possible plant condition which can be practically addressed irrespective of the probability of the occurrence of these conditions. Some of the actions for events that are beyond the design basis accident involve operations outside the control room, however these actions are also optional. The possibility that an action cannot be performed is accounted for within the procedures. The following actions outside the control room may be considered for events beyond the design basis:

1. ACTION: depressurizating the scram air header
   PURPOSE: to rapidly insert control rods in the event of multiple RPS electrical malfunctions.
   REFERENCE: T-101 Rev. G step RC/Q-12
   AREAS OUTSIDE CONTROL ROOM: R.B. Elev. 135'
   ANALYSIS: A. The procedures provides alternate methods of rod insertion such as individual rod scrams and manual rod insertion.
   B. The procedure provides guidance for plant shutdown with no rod insertion

2. ACTION: venting area above CRD pistons
   PURPOSE: to insert control rods in the event of multiple RPS electrical malfunctions
   REFERENCE: T-101 Rev. G step RC/Q-37
   AREAS OUTSIDE THE CONTROL ROOM: R.B. Elev. 135'
   ANALYSIS:
   A. The procedures provide alternate methods of rod insertion such as individual rod scrams and manual rod insertion.
   B. The procedure provides a guidance for plant shutdown with no rod insertion.

3. ACTION: venting the drywell
   PURPOSE: to maintain drywell pressure below the containment yield pressure in the event of multiple failures.
   REFERENCE: T-102 Rev. G step DW/P-19
   AREAS OUTSIDE THE CONTROL ROOM: Option 2 - Cable Spreading Room
   Option 3 - R.B. Elev. 195'
   ANALYSIS: Three venting options are available, one of which can be done entirely from the control room.

4. ACTION: injecting boron with systems other than SLC
   PURPOSE: to inject boron into the reactor in the event of multiple failures on the standby liquid control system.
   REFERENCE: T-101 Rev. G. step RC/Q-53
   AREAS OUTSIDE THE CONTROL ROOM:
   Option 1 - R.B. 195'
   R.B. 165'
   R.B. 135'
   R.B. 116'
   Option 2 - R.B. 116'
   R.B. 135'
   R.B. 195'
   Option 3 - R.B. 180'
   R.B. 165'
   ANALYSIS: Rod insertion and reactor level can be used to control reactivity until boron can be injected.

The inclusion of actions in the procedures which address emergencies beyond the design basis was in no way intended to imply that these emergencies are probable. Rather they were included as a prudent measure to provide the operator with some preanalyzed methods for maintaining the plant in a safe stable condition within the existing plant design. Since the probability of ever performing these beyond design basis actions is extremely low and because alternative actions are available for all procedure steps discussed, the areas associated with these emergency procedure steps should not be considered vital.