1881 - 1981

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June 29, 1981

RE: Docket Nos. 50-277
50-278

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Information Requested by NUREG 0737, Clarification of TMI Action Plan Requirements"

Dear Mr. Eisenhut:

Several of the TMI related requirements identified in NUREG 0737 requested the licensee to submit information regarding plans for their implementation and the results of requested engineering evaluations. A response to the following NUREG 0737 tasks is presented in the attachments. The number in parenthesis corresponds with the TMI Action Plan identification numbers.

Attachment A - Reactor Coolant System Vents (II.B.1)

Attachment B - Performance Testing of Reactor Relief and Safety Valves (II.D.1)

Attachment C - Confirm Adequacy of Space Cooling for HPCI and RCIC (II.K.3.24)

Attachment D - Effects of Loss of Alternating - Current Power on Pump Seals (II.K.3.25)

Attachment E - Improving Licensee Emergency Preparedness (III.A.2)

Drawings To: BC Aperture Card Dist.

A044 S/1

NUREG 0737, Section II.E.4.1, Dedicated Hydrogen Penetrations, requests the licensees to inform the NRC when the required design modifications have been completed. The modifications were completed on Peach Bottom Unit 2 during a refueling outage in the spring of 1980, and on Unit 3 during the refueling outage started on March 7, 1981. The modifications consisted of the installation of additional containment isolation valves to ensure that all containment purge and vent lines are single failure proof during the operation of the Containment Atmospheric Dilution system. These modifications were described in the following submittals:

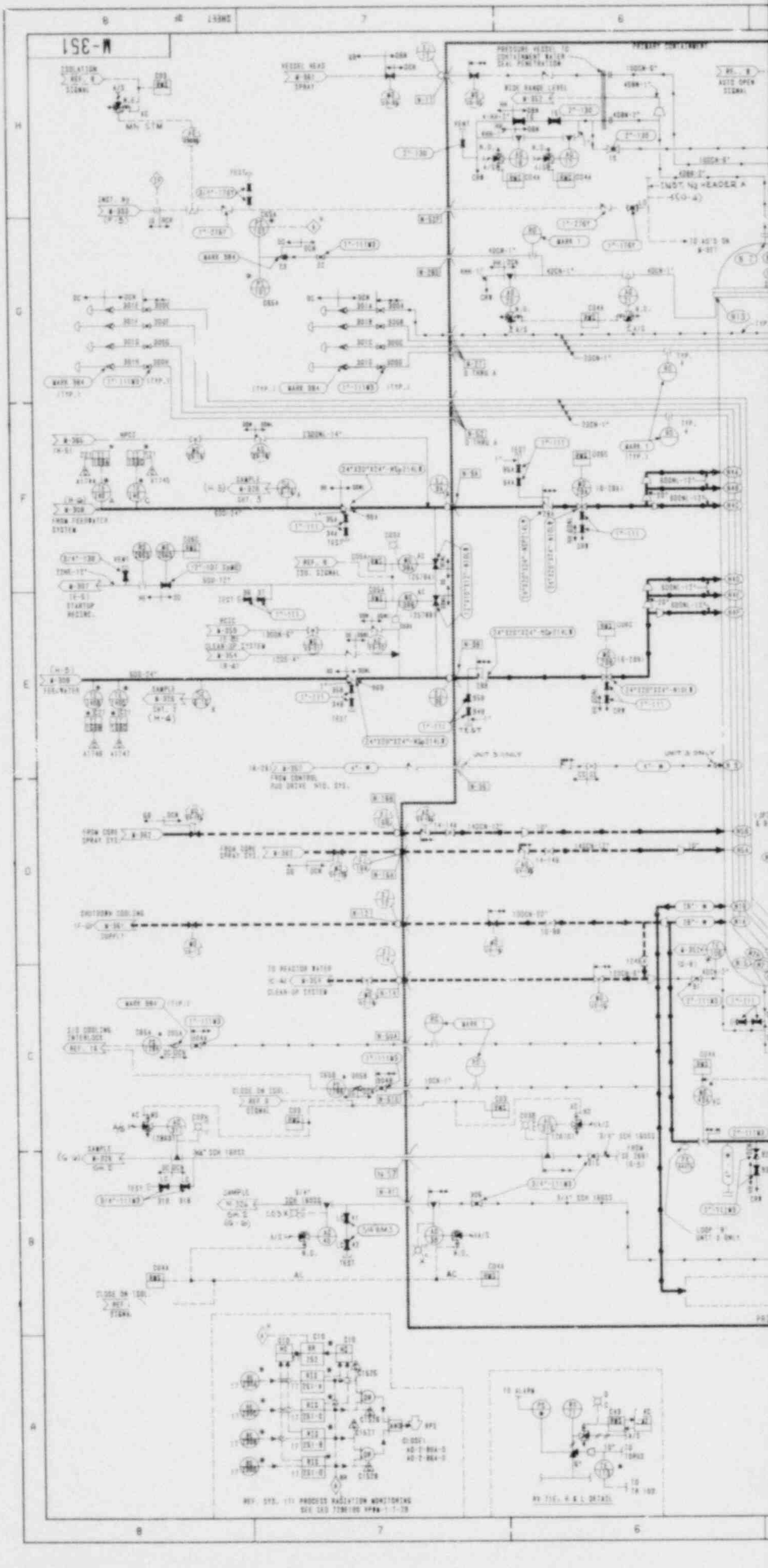
1. S. L. Daltroff, Philadelphia Electric Company, to H. R. Denton, NRC, dated January 2, 1980.

2. Application for Amendment of Facility Operating Licenses, DPR-44 and DPR-56, dated July 16, 1980.

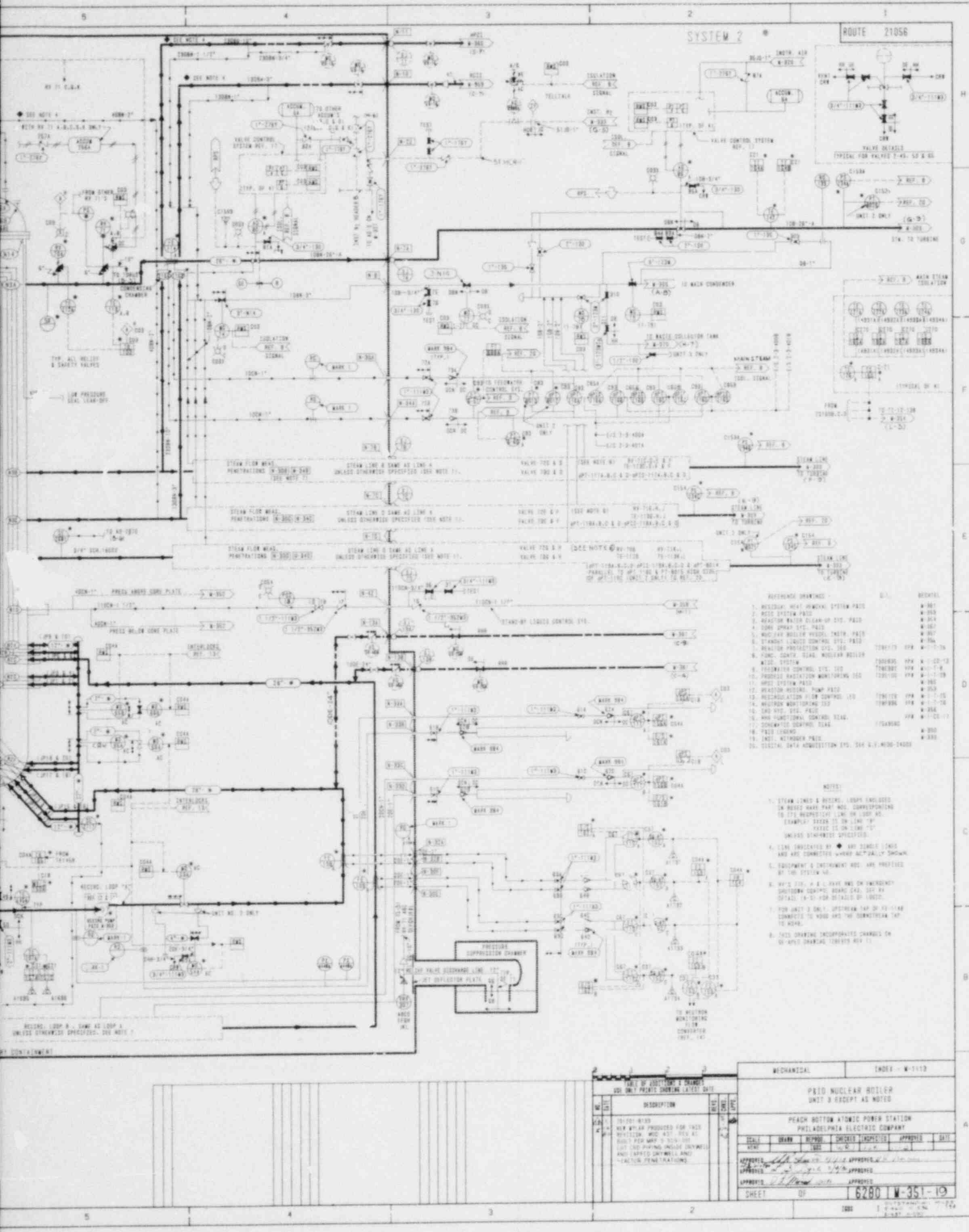
Should you have any questions regarding this submittal, please do not hesitate to contact us.

Very truly yours,

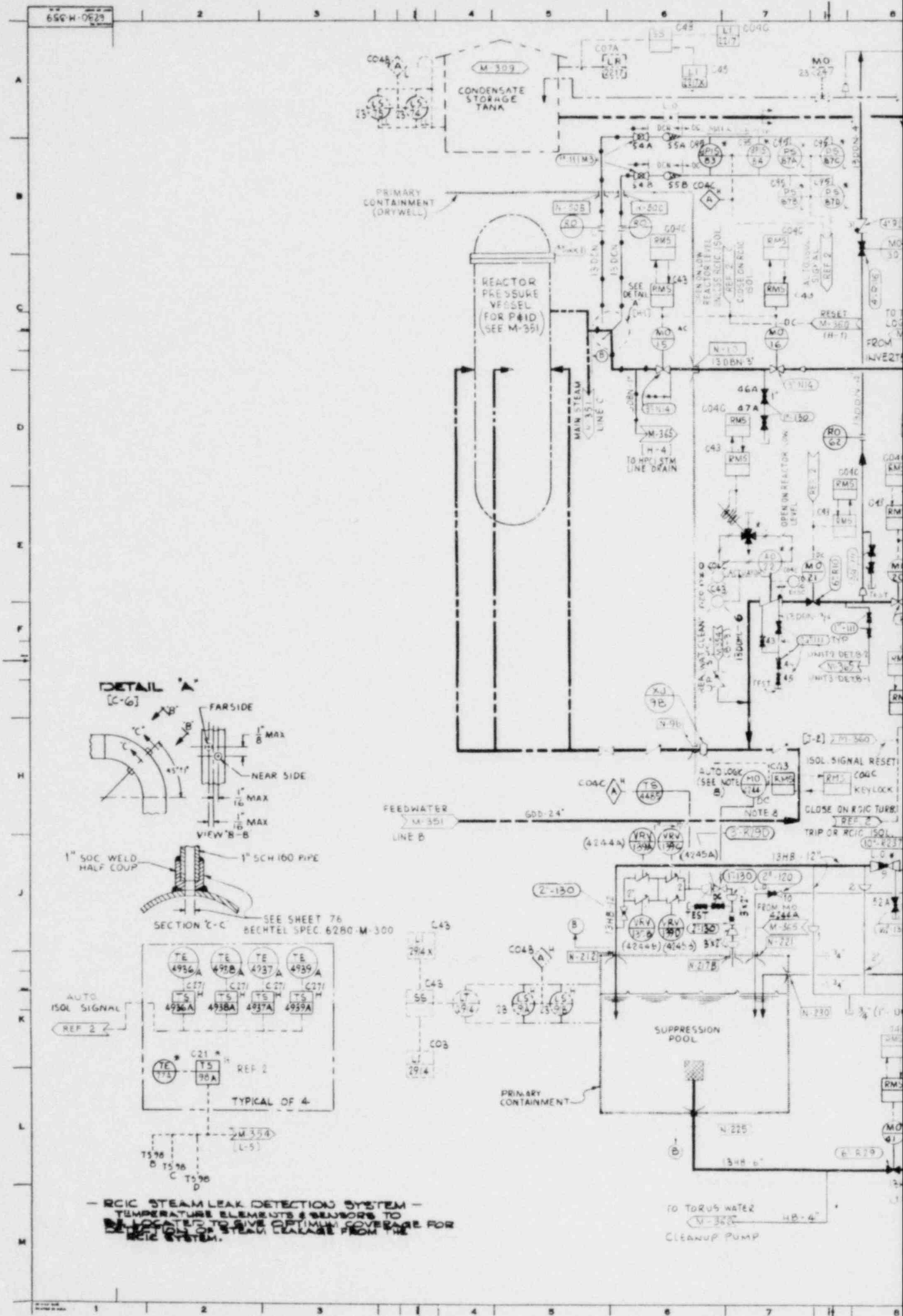
Attachments



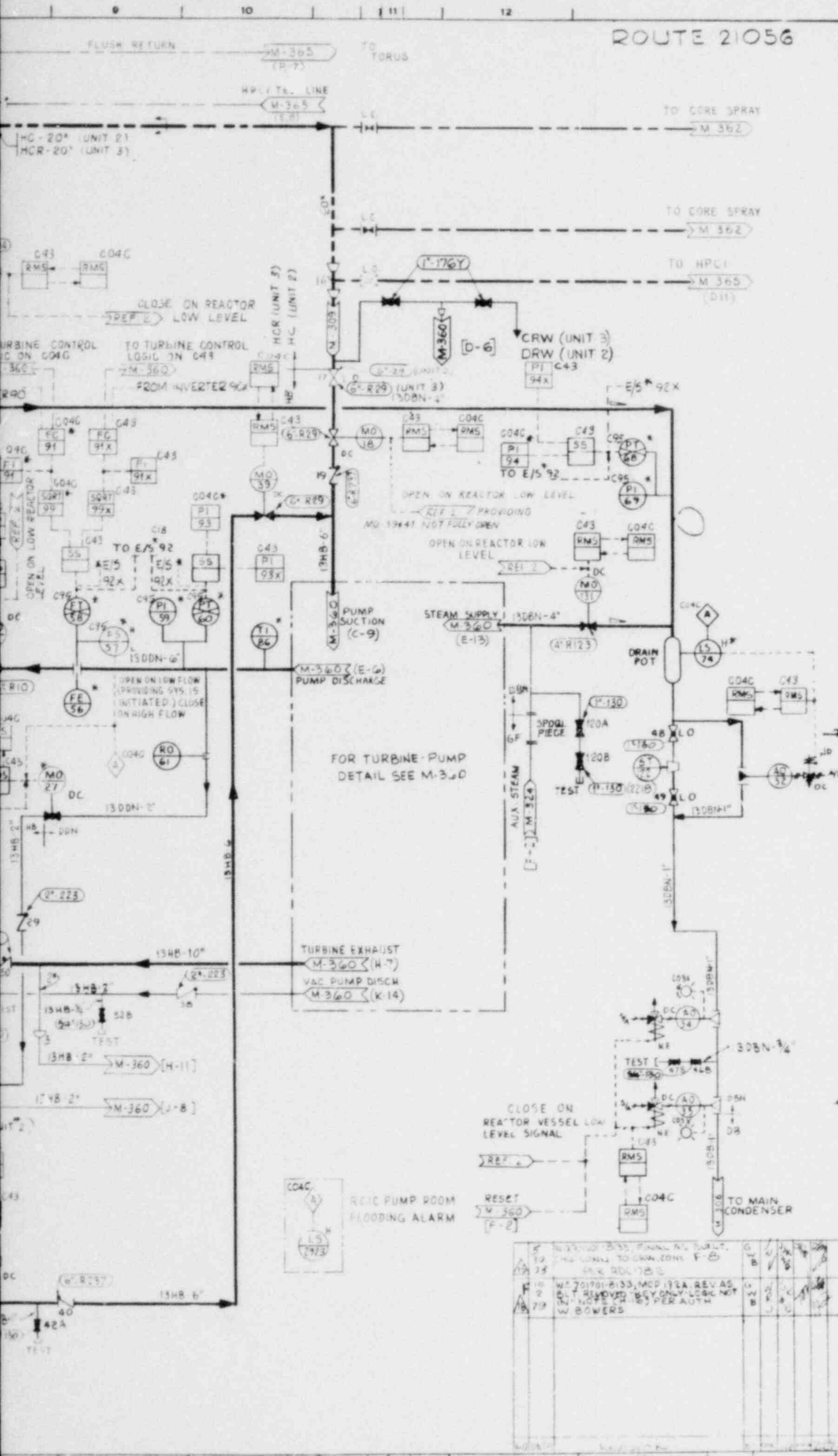
M-351
PRIMARY CONTAINMENT
FROM FEEDWATER SYSTEM



SYSTEM 2
ROUTE 21056
VALVE DETAILS
STM. TO TURBINE
MAIN STEAM ISOLATION
STEAM LINE
TO TURBINE
STAND-BY LIQUID CONTROL SYS.
PRESSURE SUPPRESSION CHAMBER
REFERENCE DRAWINGS
NOTES:
MECHANICAL
INDEX - M-1112
P&ID NUCLEAR BOILER
UNIT 3 EXCEPT AS NOTED
PEACH BOTTOM ATOMIC POWER STATION
PHILADELPHIA ELECTRIC COMPANY
SCALE
DRAWN
REPROD.
CHECKED
INSPECTED
APPROVED
DATE
NONE
APPROVED
SHEET
OF
6280
M-351-19



CONDENSATE STORAGE TANK
M-309
PRIMARY CONTAINMENT (DRYWELL)
REACTOR PRESSURE VESSEL (FOR P&ID SEE M-351)
MAIN STEAM M-351 LINE C
TO HPCI STM LINE DRAIN
FEEDWATER M-351 LINE B
SUPPRESSION POOL
PRIMARY CONTAINMENT
TO TORUS WATER CLEANUP PUMP
DETAIL "A" [C-6]
FARSIDE
NEAR SIDE
VIEW "B-B"
1" SOC WELD HALF COUP
1" SCH 160 PIPE
SECTION "C-C"
SEE SHEET 76 BECHTEL SPEC. 6280-M-300
AUTO ISOL SIGNAL
REF 2
TYPICAL OF 4
AUTO LOGIC (SEE NOTE 8)
NOTE 8
ISOL. SIGNAL RESET
CLOSE ON RCIC TURB
TRIP OR RCIC ISOL
RESET
— RCIC STEAM LEAK DETECTION SYSTEM — TEMPERATURE ELEMENTS & SENSORS TO BE LOCATED TO GIVE OPTIMUM COVERAGE FOR DETECTION OF STEAM LEAKAGE FROM THE RCIC SYSTEM.



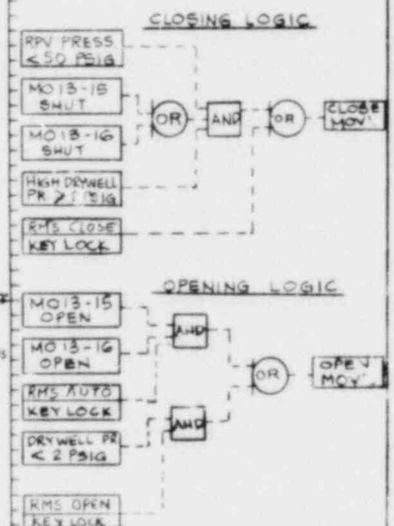
SYSTEM 13

REFERENCE DRAWINGS	BECHTEL	GE
1. NUCLEAR BOILER P&ID	M 351	728E929
2. FUNCTIONAL CONTROL DIAG. RCIC SYS.	M-1CC-33	731E307
3. NUCLEAR BOILER FUNCTIONAL CONTROL DIAG.	M-1-CC-13	730E483
4. PIPING & INSTRUMENT SYMBOLS	M 300	—
5. TURBINE CONTROL DIAGR.	M-1-G-13	2059-4
6. RADWASTE SYSTEM P&ID	M 370	728E935
7. RESIDUAL HEAT REMOVAL SYSTEM P&ID	M 361	728E930
8. REACTOR WATER CLEAN-UP SYSTEM	M 354	728E932
9. HIGH PRESSURE COOLANT INJECTION SYS. P&ID	M 365	728E936
10. NUCLEAR BOILER VESSEL INSTRUMENTATION DIAGRAM	M 352	729E426
11. PARTS LIST FOR 728E933	M-1-X-26	PL728E933
12. RCIC TURB. OUTLINE DWG.	M-1-G-3	2059-1
13. CORE SPRAY P&ID	M-362	731E384

SEE M-360 FOR NOTES 1-7

NOTES.

8. THE OPERATING LOGIC FOR MOV SHALL BE AS FOLLOWS.



THIS DRAWING INCORPORATES CHANGES ON GE DRAWING 728E933 REV. 9

F	10/2/79	WO 701701-8133, MCP 192A, REV AS BLT. REMOVED "KEY ONLY" LOGIC NOT NO. 8 (E-16) PER AUTH. W. BOWERS	G/W/B	
E	10/30/78	WO 701701-8133, FINAL AS BUILT, CHG CONN. TO CKN. CONE F-8 PER RDC 78-B	G/B	
		WO 710001-8133 MOD 281 AS BUILT PER ELEC PROD SURVEY 10-20-76		
		FINAL AS-BUILT		
		REVISED PER REV. SHT		
		REVISED PER REV. SHT		
		REVISED PER REV. SHT		
		REVISED PER REV. SHT	E.S.	
		REVISED PER REV. SHT		
		REVISED PER REV. SHT		
		REVISED PER REV. SHT		
5	5/2/68	REVISED PER REV. SHT	EH	
4		REVISED PER REV. SHT	EH	
		REVISED PER REV. SHT		
		REVISED PER REVISION NOTES		
		REVISED & REISSUED FOR INFORMATION		
		ISSUED FOR CONSTRUCTION		
		REVISIONS		

SCALE: NONE

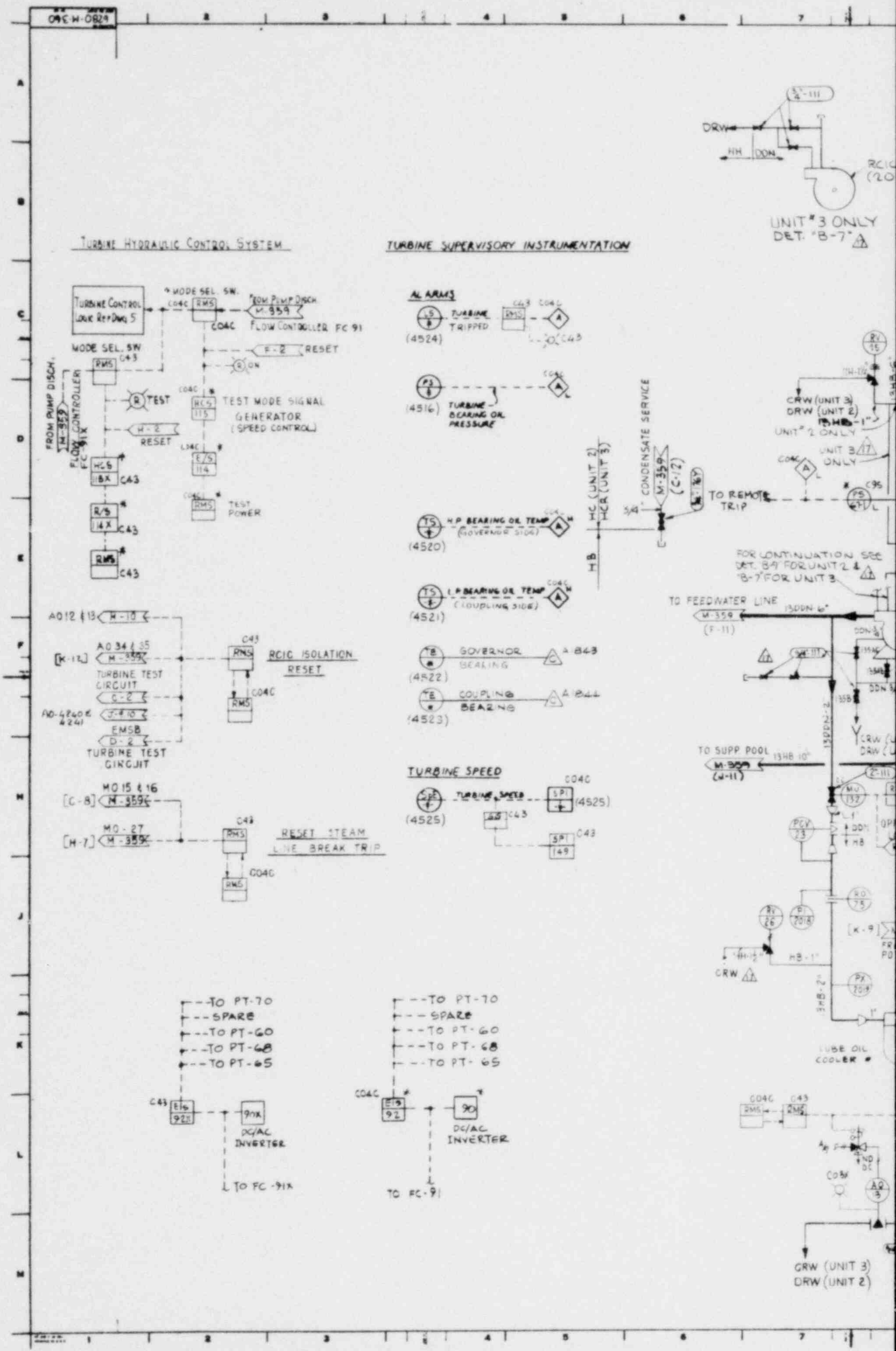
BECHTEL
SAN FRANCISCO

PEACH BOTTOM ATOMIC POWER STATION UNITS No. 2 & 3
PHILADELPHIA ELECTRIC COMPANY

P&ID REACTOR CORE ISOLATION COOLING SYSTEM

JOB No.	DRAWING No.	REV.
6280	M-359	16

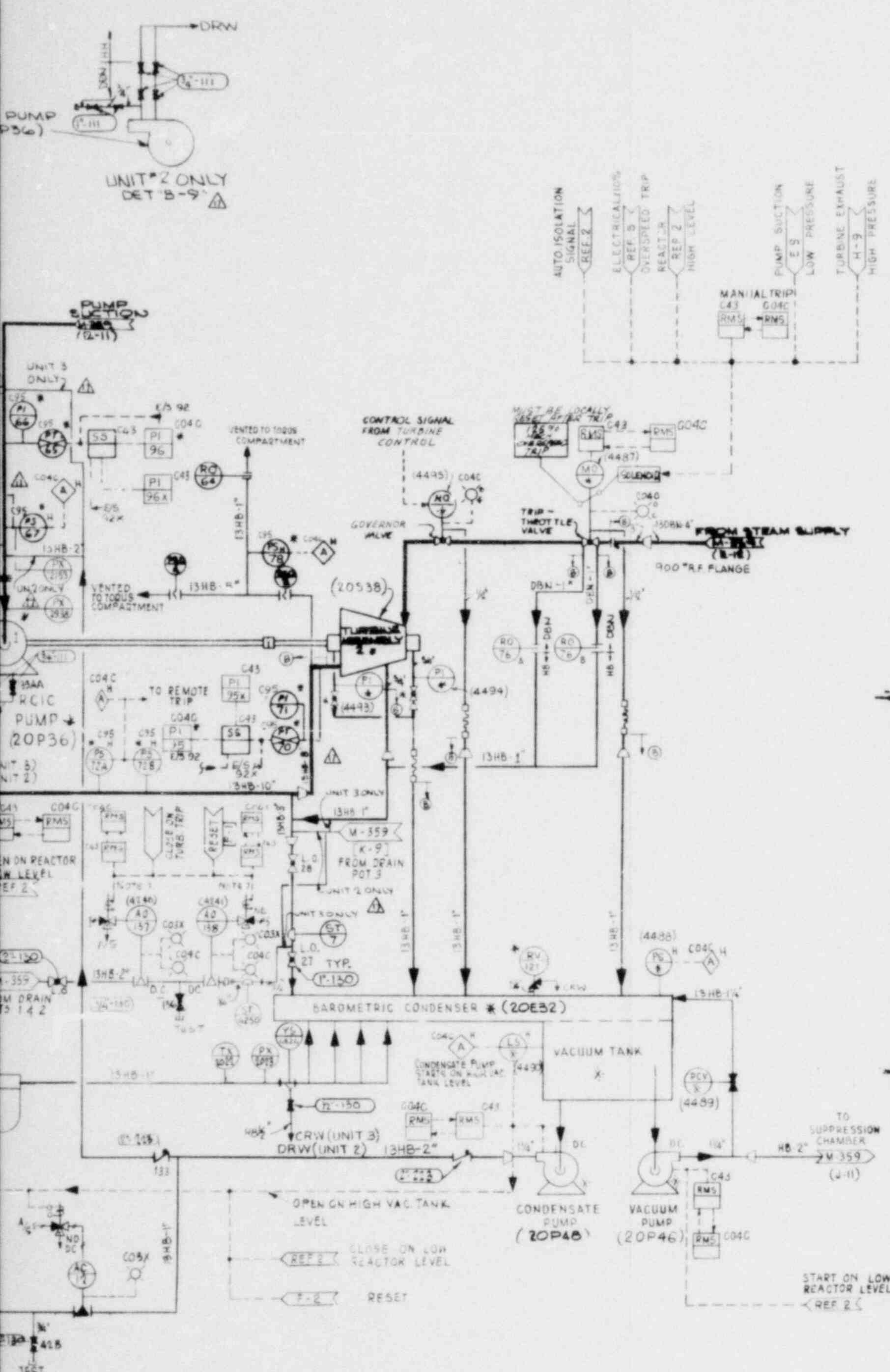
OUTSTANDING MCD



09E-H-0829
TURBINE HYDRAULIC CONTROL SYSTEM
TURBINE SUPERVISORY INSTRUMENTATION
TURBINE CONTROL LOGIC REF DWG 5
MODE SEL. SW.
FROM PUMP DISCH
M-359
FLOW CONTROLLER FC 91
RESET
ON
FROM PUMP DISCH. FLOW CONTROLLER FC 91X
TEST
TEST MODE SIGNAL GENERATOR (SPEED CONTROL)
TEST POWER
ALARMS
TURBINE TRIPPED
(4524)
(4516)
TURBINE BEARING OIL PRESSURE
H.P. BEARING OIL TEMP (GOVERNOR SIDE)
(4520)
L.P. BEARING OIL TEMP (COUPLING SIDE)
(4521)
GOVERNOR BEARING
(4522)
COUPLING BEARING
(4523)
TURBINE SPEED
(4525)
CONDENSATE SERVICE
HC (UNIT 2)
HCA (UNIT 3)
TO REMOTE TRIP
FOR CONTINUATION SEE DET. "B-9" FOR UNIT 2 & "B-7" FOR UNIT 3
TO FEEDWATER LINE
M-359 (F-11)
TO SUPP POOL
M-359 (J-11)
A012 & 13
A0 34 & 35
TURBINE TEST CIRCUIT
RCIC ISOLATION RESET
A0-4240 & 4241
EMSB
TURBINE TEST CIRCUIT
MO 15 & 16
MO-27
RESET STEAM LINE BREAK TRIP
UNIT #3 ONLY DET. "B-7"
CRW (UNIT 3)
DRW (UNIT 2)
UNIT 2 ONLY
UNIT 3 ONLY
TO PT-70
SPARE
TO PT-60
TO PT-68
TO PT-65
DC/AC INVERTER
TO FC-91X
TO FC-91
LUBE OIL COOLER
CRW (UNIT 3)
DRW (UNIT 2)

ROUTE 2105PID-B

SYSTEM 13



NOTES:

1. THE BAROMETRIC CONDENSER & VACUUM TANK SHALL BE LOCATED SUCH THAT ITS WATER LEVEL IS BELOW THE BOTTOM OF TURBINE EXHAUST FLANGE.
2. INSTRUMENT RACK FOR LOCAL INSTRUMENTATION AND TRANSMITTERS TO BE LOCATED IN SHIELDED AREA FOR ACCESS DURING TEST OPERATION.
3. SEE M-369 FOR DWG REFERENCES.
4. ALL EQUIP. & INSTR. NOS ARE PREFIXED BY THE SYSTEM NO.
5. INSTRUMENT NOS ENDING IN X (E.G. PI-96X) CORRESPOND TO INSTRUMENTS IN THE MAIN CONTROL ROOM AND ARE LOCATED ON PANEL C43 EMERG. SHUTDOWN CONTROL BOARD.
6.
7. RMS IS OF TWO POSITION "CLOSE-OPEN". VALVES OPEN IF RMS IS IN "OPEN" POSITION UNLESS TURBINE TRIP SIGNAL PRESENT.

THIS DRAWING INCORPORATES CHANGES ON GE DRAWING 728E933 SH.2 Rev. 7

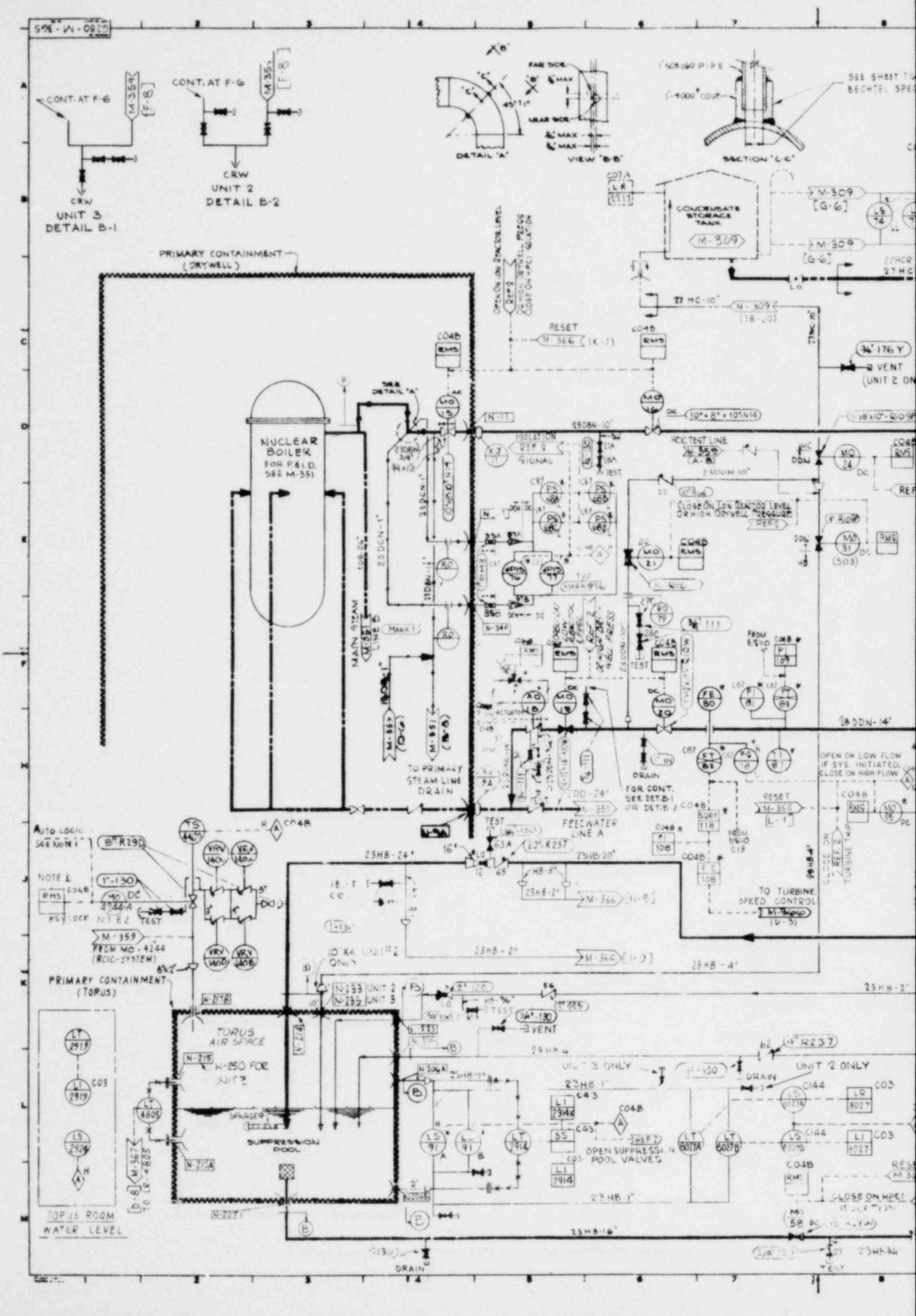
No.	Date	Revisions
17	[illegible]	W.O. 74001-8133 AS BUILT PER ELECT. PROD. SURVEY MOD 231, 10-20-[illegible]
16	[illegible]	FINAL AS-BUILT
15	[illegible]	REVISED PER REV SHT.
14	[illegible]	REVISED PER REV SHT.
13	[illegible]	REVISED PER REV SHT.
12	[illegible]	REVISED PER REV SHT.
11	[illegible]	REVISED PER REV. SHT.
10	[illegible]	REVISED PER REV SHT.
9	[illegible]	REVISED PER REV. SHT.
8	[illegible]	REVISED PER REV. SHT.
7	[illegible]	REVISED PER REV. SHT.
6	[illegible]	REVISED PER REV. NOTES
5	[illegible]	REVISED PER REV. NOTES
4	[illegible]	REVISED PER REV. SHT.
3	[illegible]	REVISED PER REVISION SHT.
2	[illegible]	REVISED PER REVISION NOTES
1	[illegible]	REVISED & REISSUED FOR CONSTRUCTION
0	[illegible]	ISSUED FOR CONSTRUCTION

BECHTEL
SAN FRANCISCO

PEACH BOTTOM ATOMIC POWER STATION UNITS No. 2 & 3
PHILADELPHIA ELECTRIC COMPANY

P&I D R.C.I.C. PUMP-TURBINE DETAILS

Job No.	Drawing No.	Rev.
6280	M-360	17


CONT. AT F-6
M-359 [F-8]
CRW
UNIT 3
DETAIL B-1
CONT. AT F-6
M-359 [F-8]
CRW
UNIT 2
DETAIL B-2
DETAIL 'A'
VIEW 'B-B'
SECTION 'C-C'
FAR SIDE
NEAR SIDE
SEE SHEET
BECHTEL SPEC
CONDENSATE STORAGE TANK
M-309
M-309 [G-6]
PRIMARY CONTAINMENT (DRYWELL)
NUCLEAR BOILER
FOR P.&I.D. SEE M-351
RESET
M-366 (K-1)
C04B
RMS
ISOLATION SIGNAL
RCIC TEST LINE
CLOSE ON LOW REACTOR LEVEL OR HIGH DRYWELL PRESSURE
23DBN-10"
23DDN-10"
27HC-10"
3/4" 176 Y
VENT
MAIN STEAM LINE D
M-351
TO PRIMARY STEAM LINE DRAIN
DRAIN
FOR CONT. SEE DET. B-1 OR DET. B-2
FEEDWATER LINE A
28DDN-14"
OPEN ON LOW FLOW IF SYS. INITIATED. CLOSE ON HIGH FLOW
RESET
M-366 [L-1]
CLOSE ON TURBINE TRIP
TO TURBINE SPEED CONTROL
FROM E5110
AUTO LOGIC SEE NOTE 1
NOTE 1
KEYLOCK NOTE 2
TEST
M-359
FROM MO-4244 (RCIC SYSTEM)
23HB-24"
23HB-20"
23HB-2"
M-366
23HB-4"
PRIMARY CONTAINMENT (TORUS)
TORUS AIR SPACE
N-250 FOR UNIT 2
SPARGER
SUPPRESSION POOL
UNIT 2
UNIT 3
VENT
UNIT 3 ONLY
UNIT 2 ONLY
DRAIN
23HB-1"
OPEN SUPPRESSION POOL VALVES
TORUS ROOM WATER LEVEL
TO LR-4803
CLOSE ON HPCI ISOLATION
23HB-16"
DRAIN
TEST

ROUTE 21056

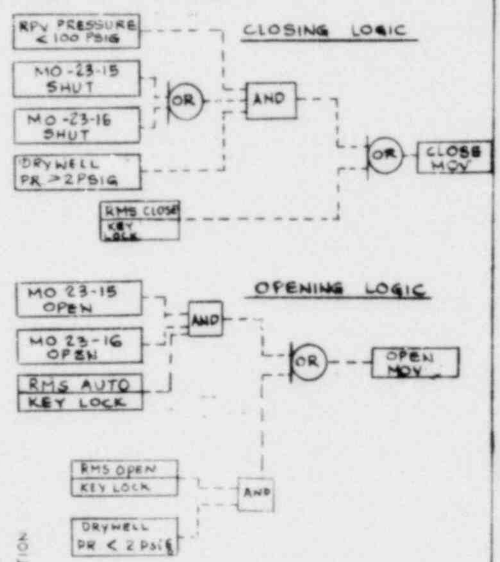
SYSTEM 23

REFERENCE DRAWINGS	GE	BECHTEL
1. NUCLEAR BOILER P&I.D.	728E929	M-351
2. FUNCTIONAL CONTROL DIAGRAM HPCI SYS.	729E5[illegible]	M-1-CC-14
3. PRIMARY CONTAINMENT SYS. ELEM. DIAG.	730E605	M-1-S-25
4. PIPING & INSTRUMENT SYMBOLS	—	M-300
5. CORE SPRAY P&I.D.	731E384	M-362
6. NUCLEAR BOILER VESSEL INST. DIAGRAM	729E426	M-352
7. RESIDUAL HEAT REMOVAL SYSTEM	728E930	M-361
8. RADWASTE SYSTEM P&I.D.	728E935	M-370
9. NUCLEAR BOILER FUNCTIONAL CONT. DIAGRAM	730E[illegible]	M-1-C-13
10. TURBINE CONTROLS DIAGRAM	2300-26	M-1-J-25
11. TURBINE OUTLINE DRAWING	2300-1	M-1-J-6
12. R.C.I.C. SYSTEM P&I.D.	728E933	M-359
13. REACTOR BLD. VENTILATION FLOW DIAGRAM	—	M-[illegible]
14. REACTOR WATER CLEAN-UP SYS. P&I.D.	729E117	M-355

SEE M-366 FOR NOTES 6-23

NOTES:

1. THE OPERATING LOGIC FOR MOV SHALL BE AS FOLLOWS.

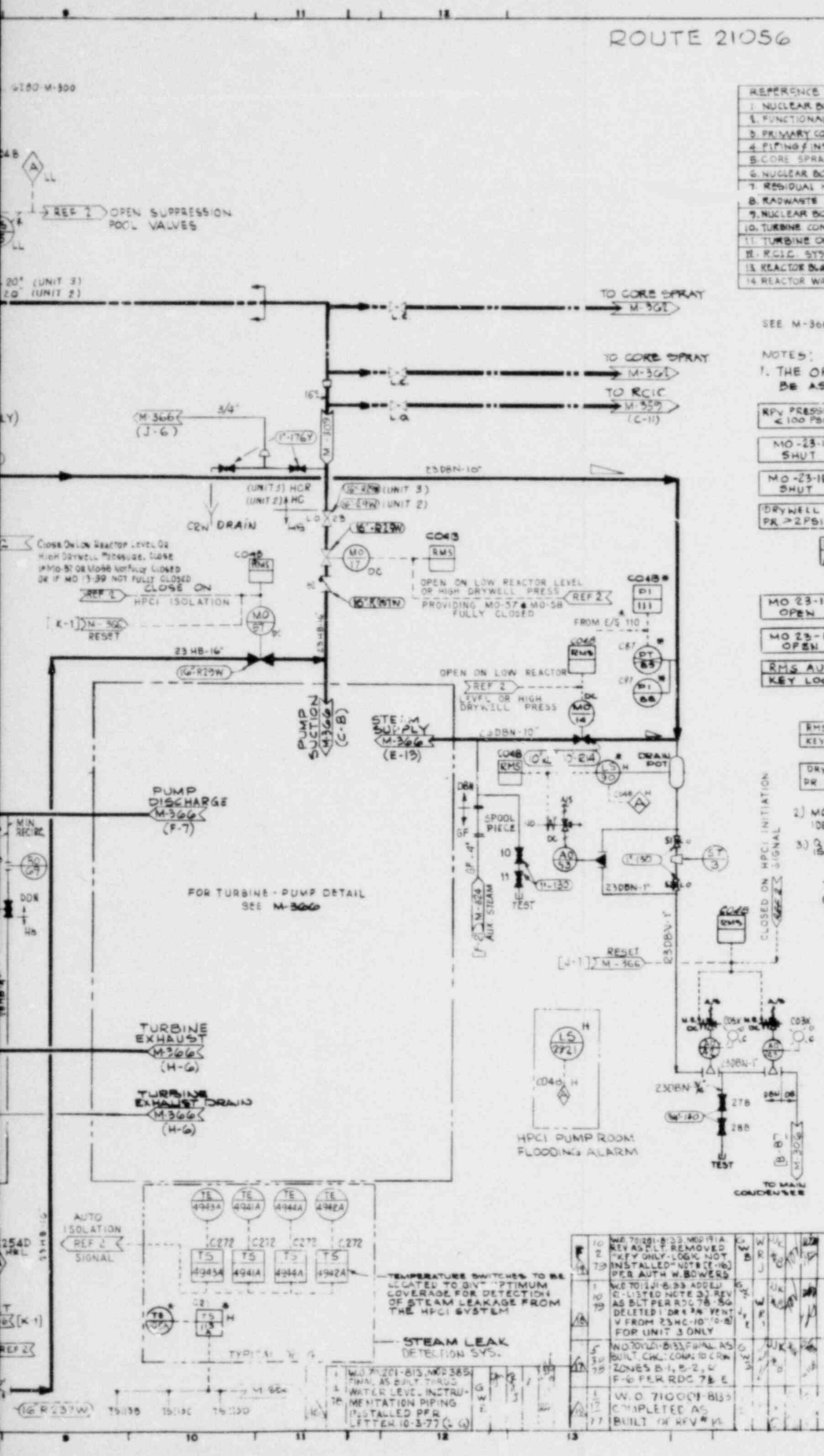


2) MO-4244 A & MO-5244A ARE ELECTRICALLY IDENTIFIED AS MO-4245 & MO-5245

3) Q-LISTED PORTION OF QAD M-465 REV 8 IS IN AGREEMENT WITH REV 18 OF THIS DWG.

THIS DRAWING INCORPORATES CHANGES OF GENERAL ELECTRIC DRAWING 728E936, SHEET 1 REV. 9

TO BE READ IN CONJUNCTION WITH M-366



No.	DATE	REVISIONS
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17	[illegible]	FINAL AS-BUILT
16	[illegible]	REVISED PER REV. SHT.
15	[illegible]	REVISED PER REV. SHT.
14	[illegible]	REVISED PER REV. SHT.
9	[illegible]	REVISED PER REV. SHT.
8	3-7-73	REVISED PER REV. SHT.
7	[illegible]	REVISED PER REV. SHT.
6	[illegible]	REVISED PER REV. SHT.
5	[illegible]	REVISED PER REV. SHT.
4	[illegible]	REVISED PER REV. NOTES
3	[illegible]	REVISED PER REV. NOTES
2	[illegible]	REVISED PER REVISION NOTES
1	[illegible]	REVISED & REISSUED FOR CONSTRUCTION
0	[illegible]	ISSUED FOR CONSTRUCTION

SCALE: NONE

BECHTEL
SAN FRANCISCO

PEACH BOTTOM ATOMIC POWER STATION UNITS No. 2 & 3
PHILADELPHIA ELECTRIC COMPANY

P&I.D. HIGH PRESSURE COOLANT INJECTION SYSTEM

JOB No.	DRAWING No.	REV.
6280	M-365	19

OUTSTANDING [illegible]

PEACH BOTTOM ATOMIC POWER STATION

Attachment A

Subject: NUREG-0737, Item II.B.1
Reactor Coolant System Vents

Refs:
a) D. B. Waters (BWROG) letter to D. G. Eisenhut (NRC) dated 4/24/81
b) D. B. Waters (BWROG) letter to D. G. Eisenhut (NRC) dated 10/8/80
c) T. D. Keenan (BWROG) letter to D. G. Eisenhut (NRC) dated 10/17/79

The BWR Owners Group position on NUREG-0737, Item II.B.1 requirements for Reactor Coolant System (RCS) Vents is contained in references a) through c). We concur with the Owners Group conclusion that adequate RCS venting capability is provided by the existing plant design. The following is a description of the existing Peach Bottom Atomic Power Station (PBAPS) provisions for RCS venting and an assessment of this capability relative to the NUREG-0737 position and clarification.

Position (1)

Peach Bottom is provided with 5 power-operated, safety-grade safety/relief valves (ADS valves RV-2-71A, B, C, G, and K) which would be the primary means of venting noncondensible gases from the reactor pressure vessel following a LOCA. The point of connection of the vent lines to the vessel is such that accumulation of gases above this elevation in the vessel will not inhibit natural circulation cooling of the reactor core. These ADS valves are self-actuating at their set relieving pressure to provide system overpressure protection and can also be actuated automatically or manually from the control room to depressurize the reactor. Operation of the ADS valves requires only safety-grade equipment and controls and does not require any source of offsite or AC power. The valves are controlled by DC power from the safeguard batteries and are pneumatically actuated from individual safety-grade accumulators. Additional information regarding the design, qualification, power source, etc. of the safety/relief valves is presented in the PBAPS FSAR Sections 4.4, 6.4, 6.5, and 7.4.

Although the power-operated, safety-grade SRV's discussed above fully satisfy the NUREG-0737 requirements, the following other means of venting noncondensibles from the reactor pressure vessel exist:

a) Six other SRV's (RV-2-71D, E, F, H, J, and L) are provided. These valves are identical to those used for ADS except that they are not equipped for automatic actuation or provided with safety-grade air supplies. They may be individually operated from the control room provided that the normal instrument air supply is available.

b) Normally closed reactor vessel head vent valves (1" - AO-2-17 and 18) are provided. These valves are operable from the control room provided that the normal instrument air supply is available. The head vent lines discharge to the drywell drain sump.

c) The main-steam driven HPCI and RCIC System turbines exhaust to the suppression pool. These will function automatically to ensure adequate core cooling and, in the process, provide continuous venting of noncondensibles to the suppression pool during their operation. The effect of noncondensibles in the HPCI and RCIC turbine steam has been analyzed and the results described in reference a).

The effects of inadvertent opening of a safety/relief valve or both head vent valves would be the same as a small steam line break. A complete steam line break is part of the plant's design basis, and smaller-size breaks have been shown to be of lesser severity. A number of reactor system blowdowns due to stuck-open relief valves have confirmed this. Similarly, a break in any of the systems enumerated above would be less severe than a complete steam line break. Since the results of the complete steam line break analysis have demonstrated compliance with the acceptance criteria of 10 CFR 50.46, no new analyses are required to show conformance with 10 CFR 50.46 for vent line failures.

Position (2)

The review and revision of PBAPS operating procedures in response to I.E. Bulletin 79-08 included plant procedures governing the operators' use of the safety/relief valves for venting the reactor pressure vessel. In the subsequent development of the BWR Emergency Procedure Guidelines (EPG) this issue was considered. The EPG regarding reactor vessel level control addresses all contingent actions required to maintain RPV level. These actions include venting for all instances when the accumulation of noncondensibles may be of concern. Further discussion of this issue is contained in reference a).

Clarification A(1)

The automatic and/or manual operation of the RCS vent paths described above provides effective venting capability to deal with large quantities of noncondensible gas. This venting capability will preclude the possibility of noncondensible gas accumulation interfering with core cooling.

Clarification A(2)

The BWR Emergency Procedure Guidelines include provisions for RCS venting for all instances when the accumulation of non-condensibles may be of concern. This topic is further discussed in reference a). As stated in the response to Position (1), the use of these vent paths or their failure will not result in a violation of the requirements of 10 CFR 50.46.

An analysis demonstrating that the direct venting of noncondensible gases into the primary containment will not result in violation of combustible gas concentration limits is presented in PBAPS FSAR Supplement 1, response to question 14.6. The gas generation rates assumed in this analysis are in accordance with 10 CFR 50.44 and Safety Guide 7 (Reg. Guide 1.7).

Clarification A(3)

Since the containments are inerted, and post-accident combustible gas control is maintained by oxygen deficiency, the PBAPS design is insensitive to the rate or extent of metal-water reaction up to the point where containment pressurization is limiting. This point is substantially beyond the present Regulatory Guide 1.7 design basis as demonstrated by the conservative assessment of this margin provided in PBAPS FSAR Section 14.6.3.3.3.

Further consideration will be given to the impact of combustible gas source terms beyond the present design basis in response to the proposed NRC degraded core rulemakings.

Clarification A(4)

As stated in the response to Position (1), a failure of one of the above vents would result in the equivalent of a small steam line break LOCA.

Clarification A(5)

SRV position indication is provided in the control room by the acoustic monitoring system previously described in response to NUREG-0578, Item 2.1.3.A (correspondence dated November 21, 1979, S. L. Daltroff, PECo. to H. R. Denton, NRC). Direct position indication is provided in the control room for the HPCI, RCIC, and head vent valves.

Clarification A(6)

Each SRV, HPCI, RCIC, and head vent valve may be individually operated from the control room.

Clarification A(7)

All design requirements for the reactor coolant system pressure boundary have been met in the design of appropriate portions of each vent path described above. Since inadvertent operation of a vent path will cause only a minor plant transient as described in the response to Position (1), redundancy is not felt to be necessary. Compliance with 10 CFR 50.46 is assured for all events as described in the response to Position (1).

Clarification A(8)

Block valves are not provided on the SRV lines in accordance with the clarification. The isolation provisions for the HPCI and RCIC steam lines will withstand a single active failure. Such provisions are not felt to be necessary for the head vent valves for the reasons discussed in the response to clarification A(7).

Clarification A(9)

The point within primary containment to which noncondensibles are vented is not of concern since the containment is inerted and effective mixing is assured. Mixing of gases within the primary containment is discussed in detail in PBAPS FSAR Supplement 1, response to question 14.6.

Clarification A(10)

The equipment, piping, controls, and position indication associated with the ADS SRV and HPCI vent paths described above have been environmentally qualified in accordance with Commission Order and Memorandum CLI-80-21 and seismically analyzed as described in PBAPS FSAR Appendix C and Supplement 1 response to questions 5.1 through 5.4 and 7.1.3. Portions of RCIC and the head vent line required for venting are also designed to withstand seismic accelerations. The RCIC v[illegible] es are environmentally qualified to achieve containment isolation.

Clarification A(11)

The SRV's and the HPCI & RCIC steam valves are tested in accordance with 10 CFR 50 Appendix J and the intent of subsection IWV of Section XI of the ASME Boiler and Pressure Vessel Code.

Clarification A(12)

This clarification does not apply since no new equipment is being provided to meet the RCS venting requirements.

Clarification B(1)

The information provided above and in the referenced letters demonstrates that the PBAPS ADS SRV's meet all of the BWR Owners Group Implementation Criteria and NRC requirements for RCS venting.

Clarification B(2)

This clarification is not applicable since PBAPS is not equipped with isolation condensers.

Copies of the following documents are submitted in accordance with NUREG-0737, Item II.B.1 - Documentation Requirement (3):

P&ID's

M-351	Nuclear Boiler
M-365	High Pressure Coolant Injection
M-366	HPCI Pump-Turbine Details
M-359	Reactor Core Isolation Cooling
M-360	RCIC Pump-Turbine Details

Schematics

M-1-S-52, Sheets 1 through 5 - Automatic Blowdown System Elementary Diagram

Test Procedures

ST 20.037 for valves MO-13-15, 16
ST 6.11

ST 20.039 for valves MO-23-15, 16
ST 6.5

ST 10.4 for valves RV-1-71A, B, C, D, E, F, G, H, J. K, L

Technical Specifications

No technical specification changes are required.

PEACH BOTTOM ATOMIC POWER STATION

ATTACHMENT B

NUREG 0737, Item II.D.1: Performance Testing of Reactor Relief and Safety Valves

Licensees shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents. Provide a submittal by July 1, 1981, confirming the adequacy of the valves based on preliminary review of generic test program results.

Response

In response to NUREG 0737, Item II.D.1, a preliminary review of the generic BWR Safety/Relief Valve (S/RV) test program (reference 1 below) has been performed regarding the adequacy of the S/RV's at the Peach Bottom Atomic Power Station for all test conditions as defined in the test description (reference 2 below). The Peach Bottom Plant employs the Target Rock, 3 stage, 6x10 type of S/RV, Model 7467F, the test results for the Target Rock model 67F test specimen are applicable to this plant's valves (reference 3 below). A preliminary review of the generic BWR S/RV test program results demonstrates that the tested valve satisfies the acceptance criteria for operability. Consequently, based on this preliminary review, the operational adequacy of the S/RV's for the Peach Bottom Plant has been demonstrated.

Ref. 1: Letter from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC), "Transmittal of Preliminary Data from Generic BWR Safety/Relief Valve (S/RV) Test Program" dated July 1, 1981.

Ref. 2: Letter from D. B. Waters (BWR Owners' Group) to R. H. Vollmer (NRC), "NUREG 0578 Requirement 2.1.2-Performance Testing of BWR and PWR Relief and Safety Valves" dated September 17, 1980.

Ref. 3: Letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC), "Responses to NRC Questions [illegible] the BWR S/RV Test Program" dated March 31, 1981 (Re[illegible] to NRC Question No. 1).

An expanded S/RV operability discussion will be submitted by October 1, 1981, as required by NUREG 0737.

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ATTACHMENT C

NUREG 0737, Item II.K.3.24: Confirm Adequacy of Space Cooling for HPCI and RCIC

This task states: "Long term operation of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Systems may require space cooling to maintain the pump room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI systems should be designed to withstand a complete loss of alternating-current power to their support systems, including coolers, for at least 2 hours".

Response

A letter, D. B. Waters (BWR Owners' Group Chairman) to D. G. Eisenhut (NRC), dated January 23, 1981, documented NRC clarification for this task. "Complete loss of alternating current power" has been clarified to mean loss of offsite power only.

At Peach Bottom the HPCI and RCIC room coolers are powered by onsite emergency power and therefore continue to be available during a loss of offsite power event. The emergency service water pumps which provide flow to the coolers are likewise powered from onsite emergency power. Adequate space cooling is therefore assured during a loss of offsite power event. There are no other supporting systems that require offsite power such that operation of the HPCI and RCIC Systems would be impaired should offsite power be lost. Therefore, modifications are not deemed to be necessary.

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ATTACHMENT D

NUREG 0737, Item II.K.3.25: Effects of Loss of Alternating Current Power on Pump Seals

This task states: "The licensees should determine on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current (a.c.) power for at least two hours. Adequacy of seal design should be demonstrated."

Response

"Loss of alternating current power" has been clarified to mean loss of offsite power only (letter, D. B. Waters, BWR Owners' Group chairman, to D. G. Eisenhut, NRC, dated January 23, 1981, documented NRC clarification).

At Peach Bottom, cooling of the recirculation pump seals is provided by two systems; the Reactor Building Cooling Water System and the Seal Purge System.

Recirculation pump vendor test data has shown that if either one of these seal cooling systems is operating, seal temperatures will remain within acceptable limits and no seal deterioration should occur. The primary cooling for the recirculation pump seals is provided by the Reactor Building Cooling Water System which cools the reactor water which flows to the lower seal cavity. After a loss of offsite power, the Reactor Building Cooling Water Pumps will restart automatically when onsite power is available. Cooling Water to the Reactor Building Cooling Water Heat Exchanger can then be established by manual realignment of the Emergency Service Water System.

Backup cooling is provided by the Seal Purge System which injects cool water from the Control Rod Drive System into the lower seal cavity. The CRD pumps are powered from the emergency diesels and can be manually restarted once onsite power is available. Hence the CRD pumps provide an alternate method that is available for seal cooling during a loss of offsite power event.

Even in the remote case where neither cooling source is reestablished and gross seal degradation were to occur, the General Electric analysis performed under the direction of the BWR Owners' Group has shown that the maximum coolant loss would be limited to 70 gpm per pump. This loss is small enough to be compensated for by normal or emergency reactor water level controls.

Various instrumentation, including seal cavity pressure, seal staging and drain flows, drywell equipment drain sump pump flow and drywell floor drain sump pump flow are available to the operator to indicate potential seal failure. In addition, gross seal failure may lead to changes in drywell pressure, temperature, or radioactivity, all of which are monitored and recorded in the control room.

During a loss of offsite power event, should the operator have an indication of gross seal failure, the leakage can be minimized by isolating the recirculation pumps. This can be accomplished by closing the suction and discharge valves, which at Peach Bottom are powered from the emergency diesels.

It is therefore concluded that a total loss of recirculation pump seal cooling is not a safety problem at Peach Bottom and modifications are not necessary.

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ATTACHMENT E

NUREG 0737, Item III.A.2: Improving Licensee Emergency Preparedness

Philadelphia Electric Company's plans for implementing Milestones 4 and 5 of NUREG 0737, Item III.A.2, are described below. The plans for implementing Milestones 1, 2, and 3 were described in correspondence dated March 31, 1981, S. L. Daltroff, Philadelphia Electric Company, to D. G. Eisenhut, NRC.

Milestone 4 - Installation of Emergency Response Facility Meteorological Hardware and Software.

Plans for upgrading the existing system which is described in our March 31, 1981, letter referenced above are as follows:

1. The Aerovane wind instruments will be replaced by lower threshold (starting speed) instruments meeting the requirements of Reg. Guide 1.23, proposed Rev. 1.

2. Turbulence class (gustiness) determinations will be supplemented by sigma theta/delta T measurements meeting the requirements of Reg. Guide 1.23, proposed Rev. 1.

3. Meteorological data will be input to a data acquisition system capable of providing 15 minute averages and 12 hours of historical data. These data will be made available for Class A dispersion model calculations, data storage, and remote interrogation per the requirements of NUREG 0654.

4. Back-up recording will be by strip chart recorders.

Milestone 5 - Full Operation of Milestone 4.

Preliminary vendor information indicates that implementation of a meteorological monitoring system modification to meet these requirements is feasible within the time frame proposed by the NRC staff. When details of the proposed modification are complete, we anticipate meeting with the NRC staff to discuss them prior to the purchase of equipment.